November 16, 2006

Mr. Dennis L. Koehl
Site Vice President
Point Beach Nuclear Plant
Nuclear Management Company, LLC
6590 Nuclear Road
Two Rivers, WI 54241-9516

SUBJECT: POINT BEACH NUCLEAR PLANT
NRC COMPONENT DESIGN BASES INSPECTION (CDBI)
INSPECTION REPORT 05000266/2006006; 05000301/2006006(DRS)

Dear Mr. Koehl:

On October 3, 2006, the U.S. Nuclear Regulatory Commission (NRC) completed a baseline inspection at your Point Beach Nuclear Plant. The enclosed report documents the inspection findings which were discussed on October 3, 2006, with you and other members of your staff.

The inspection examined activities conducted under your license, as they relate to safety, and to compliance with the Commission's rules and regulations, and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel. Specifically, this inspection focused on the design of components that are risk significant, and have low design margin.

Based on the results of this inspection, six NRC-identified finding of very low safety significance, which involved violations of NRC requirements were identified. However, because these violations were of very low safety significance, and because they were entered into your corrective action program, the NRC is treating the issues as Non-Cited Violations in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

If you contest the subject or severity of a Non-Cited Violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Point Beach Nuclear Plant.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any), will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Mark Satorius, Director
Division of Reactor Projects

Docket Nos. 50-266; 50-301
License Nos. DPR-24; DPR-27

Enclosure: Inspection Report 05000266/2006006; 05000301/2006006(DRS)
w/Attachment: Supplemental Information

cc w/encl: F. Kuester, President and Chief
Executive Officer, We Generation
D. Cooper, Senior Vice President, Group Operations
J. McCarthy, Site Director of Operations
D. Weaver, Nuclear Asset Manager
Plant Manager
Regulatory Affairs Manager
Training Manager
Site Assessment Manager
Site Engineering Director
Emergency Planning Manager
J. Rogoff, Vice President, Counsel and Secretary
K. Duveneck, Town Chairman
Town of Two Creeks
Chairperson
Public Service Commission of Wisconsin
J. Kitsembel, Electric Division
Public Service Commission of Wisconsin
State Liaison Officer

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-266; 50-301
License Nos.	DPR-24; DPR-27
Report No:	05000266/2006006; 05000301/2006006(DRS)
Licensee:	Nuclear Management Company, LLC
Facility:	Point Beach Nuclear Plant (PBNP)
Location:	Two Rivers, Wisconsin
Dates:	August 14, 2006, through October 3, 2006
Inspectors:	R. Daley, Senior Reactor Engineer, Lead Inspector A. Dunlop, Senior Reactor Engineer B. Jose, Reactor Engineer L. Kozak, Senior Reactor Analyst N. Valos, Senior Operations Inspector G. Skinner, Electrical Contractor M. Yeminy, Mechanical Contractor
Observer:	J. Bozga, Engineering Inspector
Approved by:	Mark Satorius, Director Division of Reactor Projects (DRP)

Enclosure

SUMMARY OF FINDINGS

IR 05000266/2006006; 05000301/2006006(DRS); 08/14/2006 - 10/03/2006; Point Beach Nuclear Plant; Component Design Bases Inspection.

The inspection was a 4-week onsite baseline inspection that focused on the design of components that are risk significant and have low design margin. The inspection was conducted by four regional engineering inspectors and two consultants. Six Green Non-Cited Violations (NCV) and four Unresolved Items (URI) were identified. The significance of most findings are indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process (SDP)." Findings for which the SDP does not apply may be Green, or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors, is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance involving electrical system short circuit studies. Specifically, the inspectors identified that the licensee failed to identify or analyze the potential consequences of faults on non-seismically protected circuits, or the potential for degradation of redundant trains due to a fault on a non-safety circuit that is routed in raceways associated with both redundant trains. As an immediate corrective action for this issue, the licensee performed an operability evaluation and determined that the breakers remained operable.

  The inspectors determined that the finding was more than minor because the failure to identify and analyze unacceptable consequences of overdutied circuit breakers could impact their safety function. The inspectors determined that the finding screened as Green because, despite the failure to properly analyze the consequences of overdutied circuit breakers, there was sufficient cable impedance to assure that loss of redundant buses due to postulated faults would not occur. (Section 1R21.3.b.1)

- Green. The inspectors identified a finding of very low safety significance associated with a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." Specifically, Emergency Diesel Generator (EDG) room exhaust fans, EDG diesel air start compressors, and additional loading caused by the EDG operating at frequencies above 60 Hertz (Hz) were not considered in the licensee's EDG loading calculation. The licensee determined that this issue was not an operability concern, because these additional loads did not cause the EDG to be overloaded during design basis accident conditions.

  The issue was more than minor because the failure to identify loads that would be supplied during an accident condition could result in eventual overloading of the EDG. The finding screened as having very low significance (Green) because the inspectors answered "no" to all five questions under the Mitigating Systems Cornerstone column of the Phase 1 worksheet. After performing a calculation to support operability, it was determined that there were conservatisms and other unnecessary loads in the EDG loading calculation that served to counteract the non-conservatisms that were identified by the inspection team resulting in the EDG not exceeding any vendor load limitations. (Section 1R21.3.b.2)

- Green. The inspectors identified a finding of very low safety significance associated with a violation of 10 CFR 50.63, "Loss of All Alternating Current Power." Specifically, the licensee never performed a calculation that evaluated the effects of loss of ventilation on the Auxiliary Feedwater Pump (AFP) room during a Station Blackout (SBO). The AFP rooms, which each house a turbine driven AFP (TDAFP), had not been evaluated for the heatup that would occur during the SBO 4 hour coping duration. In response to the inspector's concerns, the licensee performed informal calculations to provide reasonable assurance that the heatup in the room during an SBO would not adversely affect the equipment.

  The issue was more than minor because the licensee had not maintained a heatup calculation for the TDAFP room that assessed the effects of heatup on safe shutdown equipment as required for station blackout. The finding screened as having very low significance (Green) because the inspectors answered "no" to all five questions under the Mitigating Systems Cornerstone column of the Phase 1 worksheet. (Section 1R21.3.b.3)

- Green. The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green) involving the useable volume in the condensate storage tank (CST). Specifically, the inspectors identified that the licensee's calculation to show that there would not be vortexing in the CST was not bounding for the station blackout scenario, which was the basis for the CST volume stated in the Technical Specifications. The licensee's corrective actions included verifying the CST contained a sufficient volume to prevent vortexing in support of a station blackout scenario, and initiated actions to perform a formal calculation and to established an administrative limit to increase the available margin from the Technical Specification limit.

  The finding was more than minor because the failure to adequately evaluate the CST vortex limit could have led to an insufficient useable volume in the CST preventing the auxiliary feedwater system from performing its function during a station blackout scenario and could have affected the mitigating systems cornerstone objective of design control. The finding was of very low safety significance based on the results of the licensee's analysis and screened as Green using the SDP Phase 1 screening worksheet. (Section 1R21.3.b.4)

- Green. The team identified a Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion XI, Test Control, having very low safety significance relating to the safety-related Containment Fan Coolers (CFC). Specifically, the licensee did not assure that the fouling factor inside the tubes was maintained above the minimum specified analytical limit to prevent boiling of Service Water inside the coolers' tubes during accident conditions. The licensee visually inspected the coolers and did not establish a specific criterion for accepting a fouling factor not lower than the established minimum of 0.0003 ft^2-hr-ºF/Btu to prevent boiling inside the tubes. As an immediate corrective action, the licensee demonstrated through an evaluation that if boiling occurred, it will occur first in the upper tubes before the condition of the water in the lower tubes will cause boiling. This would result in excess service water flow to the lower tubes such that the fan coolers could still perform their safety function.

This finding was greater than minor because the current method of testing the fan coolers did not demonstrate that the existing fouling was such to prevent boiling. The finding screened as Green because, the inspectors answered “no” to the questions in the Mitigating Systems column of the SDP Phase 1 worksheet. (Section 1R21.3.b.5)

- Green. The team identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” having very low safety significance involving a modification that upgraded the Reactor Water Storage Tank/Spent Fuel Pool recirculation loop small bore piping and the Units 1 and 2 Reactor Water Storage Tank cross connect branches from the loop to Seismic Class I piping. Specifically, the inspection team found numerous non-conservative technical errors and calculation omissions in seismic design basis analysis calculations that supported this modification. This issue was entered into the licensee’s corrective action system.

  The issue was more than minor because the presence of these non-conservative calculational deficiencies resulted in seismic design basis analysis calculations to be re-performed to assure that the pipe supports would function as required during the design basis seismic event. The finding screened as having very low significance (Green) because the inspectors answered “no” to all five questions under the Mitigating Systems Cornerstone column of the Phase 1 worksheet. Specifically, after re-performing the calculations for the supports that were called into question by the inspection team, the licensee was able to show that enough margin was still available to support the loads that would be seen during the design basis seismic event. (Section 1R21.3.b.6)

B. Licensee-Identified Violations

No findings of significance were identified.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

1R21 Component Design Bases Inspection (71111.21)

.1 Introduction

The objective of the component design bases inspection is to verify that design bases have been correctly implemented for the selected risk significant components and that operating procedures and operator actions are consistent with design and licensing bases. As plants age, their design bases may be difficult to determine, and an important design feature may be altered or disabled during a modification. The Probabilistic Risk Assessment (PRA) model assumes the capability of safety systems and components to perform their intended safety function successfully. This inspectible area verifies aspects of the initiating events, mitigating systems, and barrier integrity cornerstones, for which there are no indicators to measure performance. Specific documents reviewed during the inspection are listed in the attachment to the report.

.2 Inspection Sample Selection Process

The inspectors selected risk significant components and operator actions for review using information contained in the licensee's PRA. The operator actions selected for review included actions taken by operators both inside and outside of the control room during postulated accident scenarios.

The inspectors performed a margin assessment and detailed review of the selected risk-significant components to verify that the design bases have been correctly implemented and maintained. This design margin assessment considered operational, maintenance, and design margin. Recent operations procedure changes as well as manual operator actions were considered for operational margin. Equipment reliability issues were also considered in the selection of components for detailed review. These included items such as failed performance test results, significant corrective action, repeated maintenance activities, maintenance rule (a)(1) status, components requiring an operability evaluation, NRC resident inspector input of problem equipment, system health reports, and the potential margin issues list. Consideration was also given to the uniqueness and complexity of the design, operating experience, and the available defense in depth margins. As practical, the inspectors performed walkdowns of the components to evaluate the as-built design and material condition. A summary of the reviews performed and the specific inspection findings identified are included in the following sections of the report.

.3 Component Design

a. Inspection Scope

The inspectors reviewed the Final Safety Analysis Report (FSAR), Technical Specifications (TS), component/system design basis documents, drawings, and other available design basis information, to determine the performance requirements of the selected components. The inspectors used applicable industry standards, such as the American Society of Mechanical Engineers (ASME) Code, and the Institute of Electrical and Electronics Engineers (IEEE) Standards, to evaluate acceptability of the systems' design. The review was to verify that the selected components would function as required and support proper operation of the associated systems. The attributes that were needed for a component to perform its required function included process medium, energy sources, control systems, operator actions, and heat removal. The attributes to verify that the component condition and tested capability were consistent with the design bases and were appropriate included installed configuration, system operation, detailed design, system testing, equipment/environmental qualification, equipment protection, component inputs/outputs, operating experience, and component degradation.

For each of the components selected, the inspectors reviewed the maintenance history, system health report, and corrective action process documents (CAPs). Walkdowns were conducted for all accessible components to assess material condition and to verify the as-built condition was consistent with the design. Other attributes reviewed were included as part of the scope for each individual component.

The components (19 samples) listed below were reviewed as part of this inspection effort:

- Emergency Diesel Generators (EDG): The inspectors reviewed the Diesel Room Heat Up calculations, assessing the validity of assumptions, design inputs, and results. The assessment included fan flow rates (one and two fans), relative humidity, temperature, pressure, flow path, room louvers, and the diesel de-rating curve provided by the diesel manufacturer. The team also reviewed calculations and drawings to determine if the size of the EDG was within equipment ratings as well as the EDG system health report to assess the system health and the maintenance rule status. The review also included the monthly and 18 month surveillance, EDG loading calculations, and technical specification requirements. The team also inspected several corrective action program documents that captured and analyzed various industry operating experience issues related to emergency diesel generators for their adequacy, applicability determination, and if affected, the appropriateness of corrective actions.

- Component Cooling Water (CCW) Pumps: The inspectors reviewed the schematic diagrams for the CCW pumps and verified the control logic and operation of the pumps under various scenarios. The inspectors also reviewed several maintenance procedures, the CCW system health report, and various analyses, procedures, and test results associated with operation of the CCW pumps under transient and accident conditions. The analyses included hydraulic performance and required flows during accident conditions. The team also inspected the degraded voltage calculations, breaker coordination calculation and the 125 volts control circuit voltage drop calculation to verify that the CCW pumps would have sufficient operating voltage during degraded grid voltage conditions, that breaker coordination existed between the power supply breaker to the CCW pumps and the upstream 480 V bus supply breaker, and sufficient 125 V control voltage was available to various components in the breaker control circuits. The inspectors also reviewed a modification (96-077) that added pipe support for a Unit 1 CCW line for its completeness and adequacy.

- Diesel Air Starting System: The capability of the air starting system to successfully start the emergency diesel generators was verified. The inspectors reviewed tank volume and pressure, and temperature limits, protection devices, air flow path, reliability of the air supply, and starting requirements for the tanks and valves. In addition, the inspectors reviewed diesel test results including the successful starts of each test, and vendor records ascertaining the capability of each diesel to successfully start.

- 4160 VAC Switchgear 1-A-05 (1-A-06): The inspectors reviewed Alternating Current (AC) load flow calculations to determine whether the 4160V system had sufficient capacity to support its required loads under worst case accident loading and grid voltage conditions. The inspectors also reviewed short circuit calculations to determine whether equipment and protective devices were applied within their ratings and 125 VDC (Voltage Direct Current) battery sizing and voltage drop calculations to determine whether adequate control voltage was available for accident and SBO scenarios. The inspectors reviewed elementary wiring diagrams for bus feeder and load breakers to determine whether system control logic was consistent with system design requirements stated in the FSAR. A modification package for undervoltage time delay was also reviewed to determine whether the final design was consistent with the plant design bases. The inspectors reviewed setpoint calculations for bus undervoltage relays to determine whether they afforded adequate voltage protection to equipment under degraded grid conditions and whether they were adequate to prevent spurious separation of the offsite power source. The team also inspected corrective action documents associated with identified deficiencies of the degraded voltage relay setpoints. In addition, the inspectors reviewed system health data and selected corrective action documents to determine whether there were any adverse equipment operating trends.

- 480V Switchgear 1B03 (1B04): The inspectors reviewed AC load flow calculations to determine whether the 480V system had sufficient capacity to support its required loads under worst case accident loading and grid voltage conditions. The team also inspected corrective action documents associated with identified deficiencies of load center transformer ratings. The inspectors reviewed short circuit calculations to determine whether equipment and protective devices were applied within their ratings. Since the licensee had identified deficiencies with respect to circuit breaker and bus short circuit ratings, the inspectors reviewed these corrective action documents as well. Reviews were also conducted of the 125VDC battery sizing and voltage drop calculations to determine whether adequate control voltage was available for accident and

SBO scenarios and of the elementary wiring diagrams for bus feeder and load breakers to determine whether system control logic was consistent with system design requirements stated in the FSAR. In addition, the inspectors reviewed system health data and selected corrective action documents to determine whether there were any adverse equipment operating trends.

- Auxiliary Feedwater (AFW) Pump: The inspectors reviewed various analyses, procedures, and test results associated with operation of the AFW pumps under transient, accident, and station blackout conditions. The analyses included hydraulic performance, net positive suction head (NPSH), minimum flow, and transfer of the suction source. The inspectors also evaluated the pump suction trip setpoint to verify that the pump would not inadvertently trip under transient conditions, nor would the time delay cause an inadvertent reset of the pump trip circuitry during a postulated condensate storage tank suction pipe break scenario (e.g., pressure spike). Inservice testing (IST) results were reviewed to verify acceptance criteria were met and performance degradation would be identified. In addition, the licensee responses and actions to Bulletin 88-04, "Potential Safety-Related Pump Loss," were reviewed to assess implementation of operating experience related to pump minimum flow requirements. The use of fire water as a backup supply for the turbine-driven AFW pumps bearings was reviewed to ensure sufficient flow would be provided and verifying the associated valves were adequately tested to perform their function. In addition, the inspectors reviewed initiation logic, and elementary wiring diagrams to determine whether system control logic for motor and turbine driven Auxiliary Feedwater Pumps was consistent with design bases. The inspectors reviewed motor feeder breaker setting calculations to determine whether motors and cables were adequately protected, and whether supply breakers coordinated with upstream breakers. The inspectors reviewed 125 VDC battery sizing and voltage drop calculations to determine the adequacy of control power for motor driven pump breakers, and motive power for 125 VDC Motor Operated Valves.

- 125 VDC Control Circuitry/Relaying: The inspectors reviewed battery sizing and voltage drop calculations to determine whether the 125VDC batteries and distribution equipment had sufficient capacity to perform their required functions during accident scenarios. In addition, the inspectors reviewed system health data and selected corrective action documents to determine whether there were any adverse equipment operating trends. Also reviewed were surveillance records including seven day pilot cell battery surveillance, station battery 92 day 12 month surveillance tests, and DC station battery charger maintenance procedures to verify whether technical specification surveillance requirements were satisfied. The inspectors reviewed the station battery individual cell charging procedure and a recently completed work order to verify that the individual cell charging was appropriately conducted. In addition, inspectors reviewed and verified the requirements for replacement and frequency of replacement of the electrolytic capacitors contained in the battery chargers.

- AFW Pump Discharge Steam Generator A Inlet Valve (1AF-4001): The inspectors reviewed the motor-operated valve (MOV) calculations, including required thrust, degraded voltage, maximum differential pressure, and valve weak link analysis, to ensure the valve was capable of functioning under design conditions. The inspectors also reviewed the control logic schematic and flow control diagrams to verify the adequacy of valve control logic design. Diagnostic and IST results were reviewed to verify acceptance criteria were met and performance degradation would be identified. Regulatory Information Summary (RIS) 2001-15, "Performance of DC-Powered Motor-Operated Valve Actuators," was reviewed to ensure it was properly evaluated and implemented as appropriate.

- Service Water (SW) Strainers: The inspectors reviewed the strainers' screen sizing to verify it would protect downstream components from plugging from debris from the lake. Preventive maintenance activities on the strainers were reviewed to verify performance degradation would be identified. A modification that replaced the differential pressure instruments was also reviewed.

- Auxiliary Building Supply Valve (SW-2816): The inspectors reviewed the MOV calculations, including required thrust, degraded voltage, maximum differential pressure, and valve weak link analysis, to ensure the valve was capable of functioning under design conditions. The inspectors reviewed the electrical schematic diagrams of the MOV and the safety injection (SI) signal interlock to verify that the MOV would close upon SI initiation. The inspectors also reviewed the MOV overload heater evaluation calculation to verify that the appropriate thermal overload heater was installed for this MOV. Diagnostic and IST results were reviewed to verify acceptance criteria were met and performance degradation would be identified.

- Condensate Storage Tank: The inspectors reviewed design calculations to ensure the CST contained sufficient volume to meet the Technical Specification requirement and to ensure vortexing would not occur prior to operators taking manual actions to lineup the AFW pumps to their safety-related source of water. Level instrumentation setpoint calculations were also reviewed.

- Nitrogen and Air Backup to AFW Valves: The inspectors reviewed the sizing calculations for the backup air accumulators and nitrogen tanks to ensure sufficient volume/pressure of safety-related air or nitrogen existed to fully stroke the AFW valves the required number of times based on the valves design function if normal air was lost. The inspectors also reviewed the air check valves and air system pressure tests to ensure actual system leakage was bounded by the design calculations.

- Containment Fan Coolers: The inspectors reviewed vendor design data to verify the design basis of the air-to-water coolers. The review included analyses addressing the maximum potential containment temperature humidity and pressure under accident conditions as well as inspection records and corrective action documents issued to identify any cooler deficiencies which could degrade performance. Inspectors also reviewed the heat removal capacity of the containment room coolers using accident parameters, maximum service water temperature, and maximum allowable fouling factor inside the coolers' tubes. The review assessed the licensee's commitments to Generic Letter (GL) 89-13 and the actions taken to satisfy the commitments. These included minimum flow requirements, inspection and cleaning frequencies, inspection acceptance criteria, and corrective maintenance. The inspectors assessed the licensee's compliance with GL 96-06 and the effect of very low fouling factor in the relatively new CFC coils and the potential exposure to boiling of service water in the coils due to low flow rate and high containment temperature. Containment air temperature basis calculations as well as technical specifications were reviewed to ensure that containment initial temperatures for accident assumptions were suitably conservative.

- Service Water Pumps: The inspectors reviewed piping and instrumentation diagrams, pump line up, pump capacities, and in-service testing. Also, the inspectors reviewed calculations related to pump head, flow, and NPSH to ensure the pumps were capable of providing their accident mitigation function. Design change history was reviewed, to assess potential component degradation and impact on design margins. The inspectors reviewed the control and power design drawings to verify the availability of both control and electrical power required for operability and to determine whether system control logic was consistent with design bases. The inspectors reviewed motor feeder breaker setting calculations to determine whether motors and cables were adequately protected, and whether supply breakers coordinated with upstream breakers. Reviews of the water supply (suction) path, including the susceptibility of plugging or inadvertent bypassing of the main Zurn strainer were also conducted. In addition, the inspectors reviewed the licensee responses and actions taken for compliance with GL 89-13, "Service Water System Problems Affecting Safety-Related Equipment."

- CCW Heat Exchangers: The team reviewed the Component Cooling Water (CCW) heat exchanger specifications and heat removal calculations to ensure that design basis heat removal requirements were met. The review included heat exchanger capacities, flow rates, fouling factors, and limiting Service Water temperatures. The team also reviewed the thermal performance testing and the analysis of test results with respect to the plant's commitment to compliance with the requirements and options presented in Generic Letter 89-13, “Service Water System Problems Affecting Safety-Related Equipment.” In addition, the inspectors reviewed the validity of analyses with respect to instrument uncertainties and accident conditions.

- Safety Injection Pumps: The inspectors reviewed analyses, procedures, and test results associated with operation of the SI pumps under normal (heat up and cool down) and accident conditions. The analyses included hydraulic performance, net positive suction head, minimum flow, and the capability to transfer suction from the normal source (Refueling Water Storage Tank) to the Residual Heat Removal (RHR) discharge piping. The inspectors reviewed piping and instrumentation diagrams, control logic and power supply, motor protection, pump line up, pump capacity, and vortexing possibilities. The inspectors also reviewed motor feeder breaker setting calculations to determine whether motors and cables were adequately protected, and whether supply breakers coordinated with upstream breakers. In addition, the inspectors reviewed the licensee responses and modifications made for compliance with Bulletin 88-04, "Potential Safety-Related Pump Loss."

- Refueling Water Storage Tank (RWST): The inspectors reviewed plant RWST calculations, drawings, and related operating procedures. The tank's volume, capacity, and setpoints with respect to SI suction were also assessed. In addition, the inspectors reviewed the low water level limit and its potential effect on NPSH limits and vortexing. Reviews of modifications and calculations associated with structural supports affecting the RWST were also performed.

- Injection Accumulators: The inspectors reviewed specifications, drawings, and setpoint calculations. The review also included the design volume range in the injection accumulators as well as the adequacy of the nitrogen supply. The pressure supplied to the accumulators was also assessed with respect to injection capabilities, quality ratings, operating range of pressure regulators, and setpoints and relief capacities of relief valves. In addition, the inspectors reviewed a plant modification involving replacement of the accumulator level transmitter.

- Containment Purge Supply and Exhaust Valves: The inspectors reviewed portions of the modifications that replaces the purge valves with blind flanges during power operations and put the purge valves back in service during refueling operations. Post-modification testing was also reviewed to ensure that it met applicable regulatory requirements. These modifications were also reviewed in relation to alternate source term allowances granted under recent regulatory amendments.

b. Findings

Six Green Non-Cited Violations and four Unresolved Items were identified.

1. Potential Common Mode Failure Mechanism Due to Overdutied Circuit Breakers

Introduction: The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” having very low safety significance (Green) involving electrical system short circuit studies. Specifically, the inspectors identified that the licensee failed to identify or analyze the potential consequences of faults on non-seismically protected circuits, or the potential for degradation of redundant trains due to a fault on a non-safety circuit that is routed in raceways associated with both redundant trains.

Description: In Calculation 2004-002, the licensee evaluated short circuit duty for 480V breakers and identified that several safety related circuit breakers were applied for faults in excess of their interrupting ratings. The overdutied breakers included all of the molded case breakers for loads supplied by redundant safeguards motor control centers (MCCs) 1B32 (2B32) and 1B42 (2B42). If a fault occurred on a load supplied by these

MCCs, the load breaker could fail to open and the MCC feeder breaker would be relied on to interrupt the fault. This would result in loss of power to the entire MCC. The licensee had initiated OPR000153 to support CAP 067167 and concluded that the buses remained operable because, based the single failure criterion, a fault was postulated on only one train and the redundant train would be available to perform the safety function.

The team reviewed OPR000153 and concluded that the analysis did not consider circumstances where multiple faults could result from a single event such as an earthquake. Both MCCs 1B32 (2B32) and 1B42 (2B42) supply power to several non-safety related loads that were located in non-Class 1 structures. Consequently a seismic event could cause faults on non-safety loads supplied by the redundant safety-related MCCs. The team was concerned that, if the load feeder breakers for the loads failed to open, redundant MCCs could lose power due to tripping of their feeder breakers. In response to the inspector's concern, the licensee performed additional preliminary calculations that showed that the impedance of non-safety cable routed within Class 1 seismic structures immediately downstream of the circuit breakers was sufficient to reduce fault currents below the interrupting rating of the circuit breakers in question. In a related concern, the inspectors also noted that the original separation criteria for the plant allowed a non-safety cable to be routed in raceways for both redundant trains. The inspectors were concerned that an uncleared fault on such a cable due to an overdutied breaker, could propagate damage to both redundant trains. In response to the inspectors' concern, the licensee performed an evaluation to show that there was sufficient cable length to reduce the fault current below the interrupting rating of the circuit breakers prior to the non-safety cable crossing into a routing point with the redundant safety related cable. The licensee initiated ARs 1047533 and 01051574 to address these issues.

Analysis: The inspectors determined that the failure to identify and analyze vulnerabilities associated with overdutied circuit breakers was a performance deficiency and a finding. The inspectors determined that the finding was more than minor in accordance with Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," because it was associated with the attribute of design control, which affected the Mitigating Systems Cornerstone objective of ensuring the availability and reliability of safety buses to respond to initiating events to prevent undesirable consequences. Specifically, the failure to identify and analyze unacceptable consequences of overdutied circuit breakers could impact their safety function.

The inspectors evaluated the finding using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," Phase 1 screening, and determined that the finding screened as Green because it was not a design issue resulting in loss of function per Part 9900, Technical Guidance, "Operability Determinations, and Functionality Assessments for Resolution of Degraded, or Nonconforming Conditions Adverse to Quality or Safety," did not represent an actual loss of a system safety function, did not result in exceeding a TS allowed outage time, and did not affect external event mitigation. The basis for this conclusion was that, despite the failure to properly analyze the consequences of overdutied circuit breakers, there was sufficient cable impedance to assure that loss of redundant buses due to postulated faults would not occur.

The inspectors did not identify a cross-cutting aspect with this finding.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," required, in part, that design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.

Contrary to the above, as of August 29, 2006, the licensee's design control measures failed to verify the adequacy of design, in that the potential for certain types of faults to exploit previously discovered circuit breaker overduty concerns, had not been recognized and analyzed. Specifically, the potential for loss of redundant 480V safety buses as a result of faults on non-safety related circuits had not been analyzed. The licensee entered the finding into their corrective action program as ARs 1047533 and 01051574. Because this violation was of very low safety significance, and it was entered into the licensee's corrective action program, this violation is being treated as a NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000266/2006006-01; NCV 05000301/2006006-01(DRS)).

2. Non-Conservative EDG Loading Calculation

Introduction: The inspectors identified a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green) involving calculational non-conservatisms associated with the EDG Loading calculation. Specifically, the licensee's calculation did not account for the opposite unit EDG room exhaust fans and the EDG air start compressors. Also, additional loads caused by the EDG operating at frequencies above 60 Hz were not considered in the calculation as well.

Description: Calculation N-91-016, "PBNP Diesel Generator Loading Analysis," provides the loads that would be present on each diesel generator following a Loss of Coolant Accident (LOCA) in one unit and a shutdown of the other unit concurrent with a Loss of Offsite Power (LOOP) to both units. During the review of this calculation, the inspectors determined that several loads were not accounted for.

While the EDG room exhaust fan loading were accounted for in the plant with a running EDG, the opposite unit's EDG exhaust fans were not taken into account. Since PBNP only credits one EDG per Engineered Safety Feature (ESF) division for both units, the EDG for only one unit would be expected to start. However, even though the opposite unit's EDG does not start, at least one of the EDG exhaust fans for that unit could still be operating, since the EDG room exhaust fans start due to room temperature (95 degrees F) independent of whether the associated EDG is running.

Additionally, in the case of the diesel generator starting air compressors, the licensee assigned a diversity factor of zero to the load since the compressor was assumed to operate for only a short period of time. While the operational time of the compressor would be short, the inspectors noted that the compressors would be operating right after the EDG started. Because this time period is during the injection phase of the LOCA and therefore the most limiting time period, the load should have been assumed thereby providing the most bounding condition for the EDG.

Finally, the EDG loading calculation was modeled based upon 60 Hz operation and did not consider operations at greater frequencies. Since operational procedures allowed operation up to 60.5 Hz, the load drawn from equipment operating at this higher frequency would be greater than that anticipated by the EDG load calculation.

Because these calculational deficiencies were non-conservative, the licensee entered the conditions in their corrective action program (AR 01049647). Eventual resolution of this issue will involve revision to the EDG loading calculation.

Analysis: The inspectors determined that this failure to account for EDG loads in the loading calculation was a performance deficiency warranting a significance determination. The issue was more than minor because it was associated with the Mitigating System Cornerstone attribute of "Design Control," and affected the cornerstone objective of ensuring the capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to identify loads that would be supplied during an accident condition could result in eventual overloading of the EDG.

The finding screened as having very low significance (Green) using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for the At-Power Situations," because the inspectors answered "no" to all five questions under the Mitigating Systems Cornerstone column of the Phase 1 worksheet. Specifically, after performing a calculation to support operability, it was determined that there were conservatisms and other loads in the EDG loading calculation that did not need to be included. This served to counteract the non-conservatisms that were identified by the inspection team resulting in the EDG not exceeding any vendor load limitations. The inspectors did not identify a cross-cutting aspect with this finding.

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, "Design Control" states, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Contrary to the above, design calculation N–91-016, Revision 2 did not contain the correct loads that would have been carried by the EDGs during a design basis accident. Specifically, the opposite unit's EDG room exhaust fans, the EDG air start compressor, and loading caused by running the EDG at higher frequencies were not accounted for in the loading calculation. Because this failure to account for EDG loads in the loading calculation was determined to be of very low safety significance and because it was entered in the licensee's corrective action program as AR 01049647, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000266/2006006-02; 05000301/2006006-02(DRS))

3. Lack of a 4 Hour SBO Coping Duration Heat-Up Calculation for the AFP Rooms

Introduction: The inspectors identified a Non-Cited Violation of 10 CFR 50.63, "Loss of all Alternating Current Power," having very low safety significance (Green) involving the failure to have a heat-up calculation for the AFP rooms. Specifically, as a dominant area of concern, the AFP rooms, which each house a TDAFP, had not been evaluated for heatup during the SBO 4 hour coping duration.

Description: For an SBO event, the plant was designated as a station with a 4 hour blackout duration capability. However, because the licensee installed 2 additional EDGs as qualified Alternate AC (AAC) sources that were available to power the shutdown buses within 10 minutes, the licensee, as per 10 CFR 50.63, was no longer required to have a full coping analysis. As per Nuclear Management and Resources Council (NUMARC) 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," section 7.2.4, a plant is not required to analyze the effects of loss of ventilation (heatup calculations) for dominant areas of concern like the TDAFP room if the AAC Source is used to power ESF ventilation systems and is available within 10 minutes. At Point Beach, the AAC sources do not provide power to the ventilation for the TDAFP room; consequently, the plant was required to evaluate the heatup effects for this room for the 4 hour SBO coping duration.

The TDAFP is located in the same room as the Motor Driven Auxiliary Feedwater Pump (MDAFP). During an SBO, heat loads would include the TDAFP, the MDAFP, steam pipe and steam drains associated with the TDAFP, and other miscellaneous electrical loads. These loads were not addressed for the 4 hour SBO coping duration. The licensee entered the condition in their corrective action program as AR 01051488.
The licensee performed informal calculations to provide reasonable assurance that the heatup in the room during an SBO would not adversely affect the equipment. The evaluation assumed that the TDAFP (the primary heat load in the room) would be secured after approximately 30 minutes of operation. While plant procedures did not require securing the TDAFP, it was reasonable to assume that the operators would do so due to rapid cooldown concerns. Permanent resolution of this issue will be complete when a formal calculation is performed reflecting the conditions in the TDAFP room for the 4 hour SBO Coping duration.

Analysis: The inspectors determined that this failure to perform a calculation that evaluated the effects of loss of ventilation on the TDAFP room was a performance deficiency warranting a significance determination. The issue was more than minor because it was associated with the Mitigating System Cornerstone attribute of "Design Control," and affected the cornerstone objective of ensuring the capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee had not maintained a heatup calculation for the TDAFP room that assessed the effects of heatup on safe shutdown equipment as required for station blackout.

The finding screened as having very low significance (Green) using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for the At-Power Situations," because the inspectors answered "no" to all five questions under the Mitigating Systems Cornerstone column of the Phase 1 worksheet. Specifically, the licensee performed informal calculations to provide reasonable assurance that the heatup in the room during an SBO would not adversely affect the equipment. The inspectors did not identify a cross-cutting aspect with this finding.

Enforcement: Title 10 CFR 50.63, "Loss of All Alternating Current Power," Paragraph (a)(2) requires, in part, that licensees provide sufficient capacity and capability to ensure the core is cooled in the event of a station blackout for the specified duration. It further requires that the capability for coping with a station blackout of specified duration shall be determined by an appropriate coping analysis. Finally it requires that licensees have the baseline assumptions, analyses, and related information used in their coping evaluations available for NRC review.

Contrary to the above, the licensee had never performed a coping evaluation/calculation that evaluated the effects of loss of ventilation on the TDAFP room. Because this failure to account for the heatup effects in the TDAFP room for the 4 hour SBO coping duration was determined to be of very low safety significance and because it was entered in the licensee's corrective action program as AR 01051488, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000266/ 2006006-03; 05000301/2006006-03(DRS))

4. <u>Condensate Storage Tank Vortexing Calculation Did Not Bound Station Blackout Scenario</u>

<u>Introduction</u>: The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance (Green) involving the useable volume in the CST. Specifically, the inspectors identified that the licensee's calculation to show that there would not be vortexing in the CST was not bounding for the station blackout scenario, which was the basis for the CST volume stated in the Technical Specifications.

<u>Description</u>: The inspectors reviewed Calculation 2003-0062, "AFW Pump NPSH Calculation and Condensate Storage Tank Required Fluid to Prevent Vortexing," whose purpose was to determine the bounding level in the CST above the centerline of the tank discharge pipe to the AFW pumps that would prevent vortexing in the tank. This calculation was to bound all scenarios as three AFW pumps were assumed to be operating, which would provide the highest flow (1280 gpm) and velocity (4.82 feet per second) in the discharge pipe. The calculation concluded that a level 2 feet above the centerline of the discharge pipe to the AFW pumps was required to prevent vortexing in the CST. Since this calculation did not assess the affects of this bounding vortex limit on useable volume in the CST (other calculations had used lower vortexing values), the inspectors questioned whether this value had been adequately assessed for all scenarios were the CST provided a source of water for the AFW system.

Technical Specification 3.7.6 required 13,000 gallons of useable volume, which correlated to a tank level of 7.5 feet. The inspectors questioned whether a station blackout scenario, which could be aligned from one CST at the minimum Technical Specification level of 7.5 feet, could be more a more limiting case for CST volume. Based on the diameter of the CST, 5.5 feet of CST level were needed to meet the 13,000 gallons (2350 gallons per foot) requirement as determined in TLB-34, "Condensate Storage Tank (T24A/B)." Since the 2 feet vortexing limit from Calculation 2003-0062 was considered an unuseable volume, the inspectors added this value to the 13,000 gallons useable volume (5.5 feet), which resulted in a value already at the Technical Specification limit of 7.5 feet. This basic calculation, however, did not take into account the CST level instrument tap location, nor instrument uncertainties.

The licensee initiated AR1047821 to address this issue. The licensee performed an evaluation using more realistic conditions associated with a station blackout. This included having only one turbine-driven AFW pump operating at 450 gpm, which reduced velocity in the AFW pipe to 1.69 feet per second. This resulted in a vortex limit of 1.3 feet above the centerline of the CST discharge pipe to the AFW pumps. The difference in height between the instrument tap and centerline of the pipe was determined to be 0.125 feet and instrument uncertainty was calculated as 0.53 feet. This resulted in a required level of 7.49 feet in the CST, which was within the Technical Specification limit of 7.5 feet, but contained almost zero margin. Based on this evaluation there was no operability concern.

Analysis: The inspectors determined that failure to assess the calculated vortex limit with respect to the Technical Specification limit of 7.5 feet for different scenarios was a performance deficiency and a finding. The inspectors determined that the finding was more than minor in accordance with IMC 0612, Appendix B, "Issue Dispositioning Screening," because it was associated with the attribute of design control, which affected the Mitigating Systems Cornerstone objective of ensuring the availability and reliability of the AFW system to respond to initiating events to prevent undesirable consequences. Specifically, the failure to ensure that there was an adequate useable volume in the CST to prevent vortexing during a station blackout scenario could have lead to air entrainment and possible pump failure, which could potentially render the AFW system incapable of performing its required function.

The inspectors evaluated the finding using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," Phase 1 screening, and determined that the finding screened as Green because it was not a design issue resulting in loss of function per Part 9900, Technical Guidance, did not represent an actual loss of a system's safety function, did not result in exceeding a TS allowed outage time, and did not affect external event mitigation. The basis for this conclusion was that despite the loss of design margin in the CST, the Technical Specification useable volume of 13,000 gallons was still available for a station blackout scenario. The inspectors did not identify a cross-cutting aspect with this finding.

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, "Design Control," required, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions.

Contrary to the above, as of August 31, 2006, the licensee failed to verify that the minimum CST limit was adequate to perform it's design function. Specifically, a calculation did not exist to ensure the 7.5 feet CST level specific in Technical Specification 3.7.6 was sufficient for a station blackout scenario. The licensee performed an evaluation to verify that the CST contained a sufficient useable volume that would prevent vortexing in support of a station blackout scenario. The licensee initiated AR1047821 to perform a formal calculation addressing the station blackout scenario and to establish an administrative limit of 8 feet in the CST to increase the available margin from the Technical Specification limit. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program, this violation is being treated as a NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000266/2006006-04; 05000301/2006006-04(DRS))

5. Unverified Fouling Factor Assumption for Containment Fan Coolers

Introduction: The team identified a Green Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XI, Test Control, relating to the safety-related Containment Fan Coolers for not assuring that the fouling factor inside the tubes was not maintained above the minimum specified analytical limit to prevent boiling of Service Water inside the coolers' tubes during accident conditions. Specifically, the licensee visually inspected the coolers and did not establish a specific criterion for accepting a fouling factor not lower than the established minimum of 0.0003 ft^2-hr-°F/Btu to prevent boiling inside the tubes.

Description: During refueling outages, the licensee performed periodic inspections of the Containment Fan Coolers to determine whether each fan cooler was capable of performing its safety function, namely, to remove sufficient heat from the containment atmosphere during an accident. The team questioned the ability of plant personnel to determine a narrow range of fouling by visual inspection. Specifically whether cooler inspection personnel were able to determine that the fouling factor inside the new coolers' tubes is above 0.0003. The licensee stated the coolers have developed a very thin layer of oxidation; however, the licensee could not ascertain that the fouling factor was greater than 0.0003 as required in design calculations. Cooler inspection personnel stated that a thermal performance testing program for the coolers was being considered for replacing the current inspect-and-clean program, and that if such test program is implemented, it will be capable of determining the specific fouling factor in the future. The team concluded that the uncertainty inherent in the inspection method could not ascertain that the minimum fouling factor to prevent boiling was not violated; however, concluded the Containment Fan Coolers were capable of performing their safety functions based on reasonable engineering judgment and supporting calculations.

Analysis: The failure to determine that the condition of the containment fan coolers were within the analytical limits established by engineering calculations was a performance deficiency and a finding. This finding was greater than minor because it was related to the equipment performance attribute of the Mitigating Systems Cornerstone and affects the objective of ensuring the availability of systems that respond to initiating events to prevent undesirable consequences. Specifically, higher containment temperatures during an accident affects not only containment pressure but also mitigating system availability for equipment located inside of containment. The current method of testing the fan coolers did not demonstrate that the existing fouling was such to prevent boiling which could result in higher containment temperature. The inspectors determined that this finding did not impact the Barrier Integrity Cornerstone.

The inspectors evaluated the finding using IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” Phase 1 screening, and determined that the finding screened as Green because it was not a design issue resulting in loss of function per Part 9900, Technical Guidance, ”Operability Determinations, and Functionality Assessments for Resolution of Degraded, or Nonconforming Conditions Adverse to Quality or Safety,” did not represent an actual loss of a system safety function, did not result in exceeding a TS allowed outage time, and did not affect external event mitigation. The licensee demonstrated through an evaluation that if boiling occurred, it will occur first in the upper tubes before the condition of the water in the lower tubes will cause boiling. This would result in excess service water flow to the lower tubes such that the fan coolers could still perform their safety function. The inspectors did not identify a cross-cutting aspect with this finding.

Enforcement: 10 CFR Part 50, Appendix B, Criterion XI, “Test Control,” states, in part, that a test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents.

Contrary to the above, the licensee did not establish specific acceptance criteria to assure that the fouling factor inside the coolers tubes is above the minimum analytical limit established by engineering analysis. Since the containment fan coolers are components of safety-related systems, the quality assurance requirements of 10 CFR Part 50 Appendix B apply. This item was entered into the licensee's corrective action program as AR 01051496. Because this violation was of very low safety significance, and it was entered into the licensee’s corrective action program, this violation is being treated as a NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000266/2006006-05; NCV 05000301/2006006-05(DRS))

6. Reactor Water Storage Tank/Spent Fuel Pool Pipe Support Calculation Deficiencies

Introduction: The inspectors identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” having very low safety significance (Green) involving the pipe support analysis performed for the Reactor Water Storage Tank/Spent Fuel Pool piping system. Specifically, the team identified numerous design basis calculational omissions and non-conservative technical errors associated with modification 98-021, "U0 Small Bore Pipe Support Upgrades for RWST and SFP Piping", which upgraded the Reactor Water Storage Tank/Spent Fuel Pool recirculation loop small bore piping and the Units 1 and 2 Reactor Water Storage Tank cross connect branches to Seismic Class I piping.

Description: The team reviewed modification 98-021 and calculations WE-100145, WE-100150, WE-200130, WE-200131, WE-300064, WE-300054, WE-300066 and WE-300067. Modification 98-021 upgraded the Reactor Water Storage Tank/Spent Fuel Pool recirculation loop small bore piping and the Units 1 and 2 Reactor Water Storage Tank cross connect branches so that the piping and supports would be able to withstand a design basis earthquake. These calculations provided the basis for Reactor Water Storage Tank/Spent Fuel Pool pipe support upgrades. The acceptance criteria of this modification stated “the loop piping must be able to withstand the effects of postulated seismic events and will remain Code compliant for design basis seismic loads and all other design basis loads.” The team noted that the design change, as implemented, failed to demonstrate Code compliance for Reactor Water Storage Tank/Spent Fuel Pool pipes supports analyzed for design basis seismic loads and all other design basis loads.

During the review of this modification and the associated calculations, the team identified numerous errors and omissions in the design calculations. These errors included unsubstantiated reductions in moment forces, non-conservative omissions of horizontal seismic acceleration and anchor bolt shear loading, calculational oversight of local web and flange stresses, and non-conservative weld size assumptions. The inspectors also found an instance where the anchor bolt loads assumed in the support calculation assumed upset condition loads as opposed to the required loading stresses under faulted load conditions (design basis earthquake) which are significantly higher. In response to these issues, the licensee initiated AR 01052043, AR 01052014, AR 01052649, AR 01050637, AR 01050640, AR 01052554, AR 01052513 and AR 01052446.

Subsequent engineering justification and calculation performed in these ARs provided reasonable assurance that these errors did not result in an operability concern. However, even though the licensee was able to provide this reasonable assurance for operability, the issues provided multiple examples of inadequate design basis calculations supporting this single modification.

Analysis: The team determined that the numerous design basis calculational omissions and non-conservative technical errors associated with modification 98-021 was a performance deficiency and a finding. The team determined that the finding was more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Dispositioning Screening," because it was associated with the Mitigating Systems attribute of design control, which affected the Mitigating System Cornerstone objective of ensuring the availability, reliability, and capability of the RWST during a Seismic Class I design basis event. Specifically, numerous non-conservative technical errors and calculation omissions resulted in seismic design basis analysis calculations to be re-performed to assure that the piping supports for the Reactor Water Storage Tank/Spent Fuel Pool recirculation loop small bore piping and the Units 1 and 2 Reactor Water Storage Tank cross connect branches from the loop would function as required during the design basis seismic event.

The finding screened as having very low significance (Green) using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for the At-Power Situations," because the inspectors answered "no" to all five questions under the Mitigating Systems Cornerstone column of the Phase 1 worksheet. Specifically, after re-performing the calculations for the supports that were called into question by the inspection team, the licensee was able to show that enough calculational margin was still available to support the loads that would be seen during a design basis seismic event.

The inspectors did not identify a cross-cutting aspect with this finding.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," required, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions.

Contrary to the above, calculations supporting Modification 98-021 for the Reactor Water Storage Tank/Spent Fuel Pool recirculation loop small bore piping and the Units 1 and 2 Reactor Water Storage Tank cross connect branches contained numerous non-conservative calculational assumptions and omissions affecting the design basis for the seismic analysis. However, because this violation was of very low safety significance and it was entered into the licensee's corrective action program (as stated above), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000266/2006006-06; NCV 05000301/2006006-06(DRS))

7. Broken Tie-Wraps

Introduction: The inspectors identified an Unresolved Item involving the breakage of plastic tie-wraps. Specifically, the inspectors identified that the current configuration of the plant may not be consistent with plant design documents due to the age related breakage of a large number of plastic tie-wraps used to fasten wires and cables.

Description: During field walkdowns, the inspectors noted a large number of broken plastic tie-wraps in several cable trays. Specifically, there were instances where cables were displaced outside of the confines of both vertical and horizontal trays. One instance was noted where several heavy power cables in a curved vertical tray were displaced from their required single layer configuration and were grouped on one side of the tray, and a length of several feet of one cable was outside the tray. The inspectors were concerned that tie-wraps installed to maintain required configurations could become ineffective as a result of embrittlement due to aging.

The inspectors noted that electrical installation specifications and design guides required securing cables for such purposes as maintaining spacing for power cable ampacity, maintaining stiffness in unsupported lengths of wire bundles to ensure minimum bending radius, and maintaining cables within vertical raceways. In addition, the General Implementation Procedure for Seismic Qualification User Guidelines (SQUG) walkdowns provided criteria for checking the integrity of tie-wraps to ensure that required configurations would not be compromised by failure of embrittled tie-wraps. Because of the large number of broken ties, it appeared that the existing plant configuration was not consistent with the snapshot evaluation performed during the SQUG walkdowns performed in the early 1990's.

Following identification of this issue, the licensee performed limited field inspections of cable trays and found four instances of cables outside the trays (the NRC inspectors subsequently found one more). The licensee also performed limited "pull tests" believed to be similar to the ones performed during the original SQUG walkdowns, and observed breakage of 6 of the 50 tie-wraps tested. These instances were documented in the licensee's corrective action program as ARs 1051182 and 01052281, but have thus far been characterized as "housekeeping" concerns because the licensee stated that tie-wraps were not required to maintain spacing for ampacity after the initial installation due to stiffness of wires, and that tie-wraps were only intended as temporary supports during construction.

The current condition of tie-wraps and the determination whether the tie-wraps are required to maintain design assumptions is unresolved pending further NRC review of the licensee's design specifications and of SQUG commitments and license basis. (URI 05000266/2006006-07; 05000301/2006006-07(DRS))

8. Safety Related Equipment Not Protected from Tornado Missiles

Introduction: The inspectors identified an Unresolved Item involving the lack of tornado missile protection for some safety related equipment. Specifically, the inspectors identified that the G-01 and G-02 diesel generator exhaust stacks were not protected from tornado missiles, and that the G-01 and G-02 diesel generator room exhaust fans could be degraded by tornado induced damage.

Description: The inspectors noted two areas of vulnerability which could result in degradation of the G-01 and G-02 diesels due to tornado missiles. In the first case, the diesel generator exhaust stacks were routed on the outside of the turbine building and were unprotected from missiles. In the second case, the diesel generator room air exhaust fan "dog houses" were located in close proximity to the CSTs and were subject to flooding resulting from a tornado induced CST failure.

The G-01 and G-02 diesel generator exhaust stacks were routed vertically upward from the diesel generator rooms, on the outside of the east wall of the turbine building. The stacks are approximately 100 ft. high and 26 inches in diameter. The inspectors discovered that the exhaust stacks had not been analyzed for the effects of tornado missiles. Consequently, the inspectors were concerned that a tornado induced missile could damage the stacks to a sufficient extent that exhaust flow would be restricted and the diesels would not be capable of performing their required function. In response to the inspectors concern, the licensee provided an informal calculation, based partially on data from the PBNP Individual Plant External Event Evaluation (IPEEE), that showed a very low probability of diesel generator exhaust stack damage from a tornado missile (1.95E-7 /yr).

The inspectors were also concerned that a tornado generated missile could rupture the CSTs, causing the release of water through the rupture, and possibly causing catastrophic failure of the tanks. The CSTs were located at the 26' elevation of the turbine building directly above the G-01 and G-02 diesel rooms. The diesel room exhaust fan "dog houses" were also located on the 26' elevation adjacent to the CSTs. The inspectors were concerned that water from a ruptured CST could enter the louvered openings of the exhaust fan doghouses, and damage the exhaust fans located directly below. The fans appeared to be susceptible to damage due to the close tolerances used in their vane-axial design. Although the CSTs have been analyzed to be capable of withstanding the effects of an earthquake, or the direct wind loading of a tornado, they have not been analyzed for the effects of tornado missiles.

In response to the inspector's questions, the licensee also identified other safety related components exposed to tornado missile damage, including the G-01 and G-02 Day Tank Flame Arrestors, and the G-01 and G-02 Room Air Exhaust Fan louvers. (The inspectors identified susceptibility of these components to damage from flooding, but the licensee identified that they also represent a small but direct tornado missile target.)

The inspectors note that FSAR Section 1.3, Table 1.3-1 includes the following requirements:

> Those systems and components of reactor facilities which are essential to the prevention or to the mitigation of the consequences of nuclear accidents which could cause undue risk to the health and safety of the public shall be designed, fabricated, and erected to performance standards that enable such systems and components to withstand, without undue risk to the health and safety of the public, the forces that might reasonably be imposed by the occurrence of an extraordinary natural phenomenon such as earthquake, tornado, flooding condition, high wind, or heavy ice. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been officially recorded for the site and the surrounding area; and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

In addition, Section 1.3.1 – Overall Plant Requirements (GDC 1- GDC-5) states, in part:

> Similarly, measures are taken in the plant design to protect against high winds, flooding, and other natural phenomena. The containments and Seismic Class I portions of the Auxiliary Building, the turbine hall, and the pumphouse are designed to withstand the effects of a tornado.

The criteria for tornado missile design for Class I *structures* is documented in Bechtel Report B-TOP-3, as a 108 lb. 4 ” x 12 ” x 12 ’ wood plank traveling 300 mph.

Neither the FSAR or the Bechtel Report define criteria for tornado missile design affecting *Systems, Components, and Non-Class 1 Structures*. The FSAR does not recognize the probabilistic approach used to justify the vulnerability of the diesel generator exhaust stacks. When probabilistic techniques are used for tornado protection at nuclear power plants, these are typically explicitly allowed by the FSAR. In response to the inspector’s concerns the licensee initiated AR 01047610 to address the lack of an explicit licensing basis for protection of systems and components against the effect of tornado missiles. This issue is unresolved pending further NRC review of the licensee’s design and license basis. (URI 05000266/2006006-08; URI 05000301/ 2006006-08(DRS))

9. <u>EDG Testing in Surveillance Requirement (SR) 3.8.1.5 not Bounding</u>

<u>Introduction</u>: The inspectors identified an Unresolved Item involving EDG testing. Specifically, the team questioned the adequacy of the testing being performed to meet SR 3.8.1.5. This issue is unresolved pending further NRC review of the plant’s licensing basis and testing methodology.

<u>Description</u>: Technical Specification SR 3.8.1.5 states, in part:

> “Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated ESF actuation signal:
>
> c. Standby emergency power source auto-starts from standby condition and:
>
> 1. energizes permanently connected loads”

The licensee has four EDGs; however, only two EDGs are credited for accident mitigation. Consequently, as per Limiting Condition for Operability (LCO) 3.8.1, only two EDGs - one for each division supporting both units - are required for Operability. During the bounding design basis accident, a LOOP/LOCA on one unit and a LOOP on the other unit, the plant would supply both units' divisional ESF buses from a single EDG.

Because of the design basis for the EDGs, the inspectors questioned whether the testing performed in SR 3.8.1.5 tested both the permanently connected loads for the LOOP/LOCA plant and the LOOP plant. The licensee did not test all permanently connected loads at one time. Instead, the permanently connected loads were tested at different time periods. As explained by the licensee, during each unit's refueling outage each divisional ESF bus is tested to both LOOP and LOOP/LOCA conditions; however, the EDGs are not tested with both the LOOP and the LOOP/LOCA permanently connected loads. During the design basis accident, both LOOP and LOOP/LOCA permanently connected loads would connect almost simultaneously. The immediate effects of these loads starting simultaneously would challenge the EDG and its control systems more than the testing that is currently being conducted. The inspectors were concerned because the loads that the EDG would have to connect to during an actual design basis accident were not being tested. The inspectors believed that this did not meet the wording in SR 3.8.1.5 nor did it meet the intent of the surveillance.

Following identification of this issue, the licensee entered the issue into their corrective action program as AR 01053357. This issue is unresolved pending further NRC review of the Point Beach testing and licensing basis for the EDGs. (URI 05000266/2006006-09; 05000301/2006006-09(DRS))

10. <u>EDG Endurance Test not Being Performed</u>

<u>Introduction</u>: The inspectors identified an Unresolved Item involving endurance and capacity testing of the EDGs. Specifically, the Technical Specifications contain no requirements to perform an endurance test. This issue is unresolved pending further NRC review of the plant's licensing basis and the safety significance of this issue.

<u>Description</u>: The licensee's Technical Specifications do not contain requirements for performing an endurance run on the EDGs. An endurance run tests the ability of the EDG to remain operationally intact for a potentially long period of time. Its primary purpose is to demonstrate that each EDG is in operational readiness to assume the design basis accident loads. A standard time period for such an endurance test would be 24 hours. The licensee's longest EDG TS surveillance (SR 3.8.1.3) is a one hour test that does not bound predicted accident condition loads.

The inspectors were concerned that without an endurance run requirement the present TS surveillances did not adequately test the EDGs to ensure that they could perform their design basis accident function. The endurance run gives confidence in the readiness of the EDG to deliver its design basis loads for an extended period by challenging the EDGs mechanical systems, electrical systems, and control systems. Without this test, the inspectors were not confident in the ability of the EDG to perform its design function.

However, since there were no apparent regulatory requirements in place for an EDG endurance run test, the inspectors were unable to apply enforcement in regard to this issue. Additionally, while the licensee has not historically performed an endurance test for the EDGs, they intended to voluntarily perform such a test during the next refueling outage for each unit's EDGs. Although the licensee intended to perform the endurance test in the near future, the inspectors were still concerned, because without a regulatory requirement in place to perform such a test, discontinuing the testing in the future would be a possibility. Additionally, without a regulatory requirement, valid acceptance criteria for the testing would be an option rather than a requirement.

This issue is unresolved pending further NRC review of the Point Beach licensing basis for the EDGs and to determine NRC courses of action for resolution in the future. (URI 05000266/2006006-10; 05000301/2006006-10(DRS))

.4 Operating Experience

a. Inspection Scope

The inspectors reviewed five operating experience issues (5 samples) to ensure these issues, either NRC generic concerns or identified at other facilities, had been adequately addressed by the licensee. The operating experience issues listed below were reviewed as part of this inspection effort:

- IE Bulletin No. 80-04, "Analysis of a PWR Main Steam Line Break With Continued Feedwater Addition";
- OE 22166, "Air Void in ECCS Recirculation Piping at Palo Verde";
- IN 2003-19, "Unanalyzed Condition of Reactor Coolant Pump Seal Leak-off Line During Postulated Fire Scenarios or Station Blackout";
- BL 88-04, "Potential Safety-Related Pump Loss"; and
- RIS 2001-015, "Performance of DC MOV Actuators."

b. Findings

No findings of significance were identified.

.5 Modifications

a. Inspection Scope

The inspectors reviewed four permanent plant modifications related to the selected risk significant components to verify that the design bases, licensing bases, and performance capability of the components have not been degraded through modifications. The design changes listed below were reviewed as part of this inspection effort:

- 94-066*A, “Soft Face Check Valve Disk (CSI-834D) and Relief Valve Installation”;

- 95-048, “U1 EDG Output Breaker Undervoltage Permissive Time Delay”;

- 01-098, “Upgrade Service Water Zurn Strainer D/P Indication and Alarm Instrumentation”; and

- TM 04-001, “Temporary Replacement of Unit 1 Purge Supply/Return Valves.”

b. Findings

No findings of significance were identified.

.6 Risk Significant Operator Actions

a. Inspection Scope

The inspectors performed a margin assessment and detailed review of five risk significant, time critical operator actions (5 samples). These actions were selected from the licensee’s PRA rankings of human action importance based on risk achievement worth values. Where possible, margins were determined by the review of the assumed design basis and FSAR response times and performance times documented by job performance measures results. For the selected operator actions, the inspectors performed a walk through of associated procedures with a plant operator to assess operator knowledge level, adequacy of procedures, and availability of special equipment where required. The following operator actions were reviewed:

- Actions to establish backup fire water supplies to the Condensate Storage Tanks;

- Actions to establish the G05 Gas Turbine as an alternate AC power source during a Station Blackout event;

- Actions to transfer the Emergency Core Cooling System (ECCS) to containment sump recirculation during a large break LOCA;

- Actions to prevent Steam Generator overfill in response to a Steam Generator Tube Rupture (SGTR); and

- Actions to isolate an Instrument Air header break.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA2 Problem Identification and Resolution

.1 Review of Condition Reports

a. Inspection Scope

The inspectors reviewed a sample of the selected component problems that were identified by the licensee and entered into the corrective action program. The inspectors reviewed these issues to verify an appropriate threshold for identifying issues and to evaluate the effectiveness of corrective actions related to design issues. In addition, corrective action documents written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problem into the corrective action program. The specific corrective action documents that were sampled and reviewed by the inspectors are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exits

.1 Exit Meeting

The inspectors presented the inspection results to Mr. D. Koehl and other members of licensee management at the conclusion of the inspection on October 3, 2006. No proprietary information was identified.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
C. Butcher, Engineering Director
K. Dittman, Electrical Design Engineering Supervisor
F. Flentje, Regulatory Affairs Supervisor
J. Golding, Emergency Diesel Generator Engineer
F. Hennessy, Programs Engineering Supervisor
K. Holt, Electrical Analysis Supervisor
T. Kendall, Senior Technical Advisor - Engineering
T. Lensmire, Electrical Analysis Engineer
J. Masterlark, PRA Supervisor
J. McCarthy, Site Director
M. Millen, Work Control Center Manager
D. Pederson, Emergency Diesel Generator Engineer
L. Peterson, Design Engineering Manager
M. Ray, Regulatory Affairs Manager
B. Schaueble, Electrical Design Engineer
T. Vandenbosch, Operations Supervisor
W. Zipp, Mechanical Systems Engineering Supervisor

Nuclear Regulatory Commission
G. Gibbs, Resident Inspector
R. Krsek, Senior Resident Inspector
A. M. Stone, Chief, Engineering Branch 2, DRS

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000266/2006006-07; 05000301/2006006-07	URI	Broken Tie-Wraps
05000266/2006006-08; 05000301/2006006-08	URI	Safety Related Equipment Not Protected from Tornado Missiles
05000266/2006006-09; 05000301/2006006-09	URI	EDG Testing in SR 3.8.1.5 not Bounding
05000266/2006006-10; 05000301/2006006-10	URI	EDG Endurance Test not Being Performed

Opened/Closed

05000266/2006006-01; 05000301/2006006-01	NCV	Potential Common Mode Failure Mechanism Due to Overdutied Circuit Breakers
05000266/2006006-02; 05000301/2006006-02	NCV	Non-Conservative EDG Loading Calculation
05000266/2006006-03; 05000301/2006006-03	NCV	Lack of a 4 Hour SBO Coping Duration Heat-Up Calculation for the AFP Rooms
05000266/2006006-04; 05000301/2006006-04	NCV	Condensate Storage Tank (CST) Vortexing Calculation Did Not Bound Station Blackout Scenario
05000266/2006006-05; 05000301/2006006-05	NCV	Unverified Fouling Factor Assumption for Containment Fan Coolers
05000266/2006006-06; 05000301/2006006-06	NCV	Reactor Water Storage Tank/Spent Fuel Pool Pipe Support Calculation Deficiencies

Discussed

None.

LIST OF DOCUMENTS REVIEWED

The following is a list of licensee documents reviewed during the inspection, including documents prepared by others for the licensee. Inclusion on this list does not imply that NRC inspectors reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document in this list does not imply NRC acceptance of the document, unless specifically stated in the inspection report.

<u>1R21</u> <u>Component Design Bases Inspection</u>

<u>Calculations</u>

292-002-001; Pump Temperature Rise and NPSH Margin Calculation; Revision 0

6704.001-C-017; Effect of Tornado Generated Missiles on EDG Building; dated December 29, 1984

6704.001-C-087; Diesel Building, Calculation of HVAC Requirements; dated May 18, 1995

692301-2.2-004-00-A; AFW Pump Room Loss of HVAC Analysis; Revision 5

91C2696-C-014; USI A-46/IPEEE Equipment Fragilities for T-24A and T-24B; Revision 0

97-0118-00-A; Capability to Achieve Cold Shutdown in Both Units with One CCW Pump and Two CCW Heat Exchangers; dated September 8, 1999

97-0118-00-B; Capability to Achieve Cold Shutdown in Both Units with One CCW Pump and Two CCW Heat Exchangers; dated May 30, 2002

97-0155; Auxiliary Feedwater Pump Low suction Pressure Trip Instrument Loop Uncertainty/Setpoint Calculation Unit 2 Operation; Revision 1

97-0172; Available Water in Volume of Piping to the Auxiliary Feedwater Pumps Following Pipe Break at Elevation 25-6; Revision 2

97-0215; Water Volume Swept by All Four AFW Pumps Following a Seismic/Tornado Event Affecting Both Units; Revision 5

97-0231; Auxiliary Feedwater Pump Low suction Pressure Trip Instrument Loop Uncertainty/Setpoint Calculation Unit 1 and Unit 2 Operation; Revision 0

98-0051; Service Water System Heat Exchanger HX-55A/B Flow Requirements; Revision 2

98-0172-02-B; Containment Fan Cooler Service Water Acceptance Criteria; dated December 30, 2003

99-0032; Application of Uncertainty to Hydraulic Modeling of the Service Water System; Revision 1

2001-0056; TDAFP Mini Recirc Valve (1/2AF-4002) Instrument Air Accumulator Sizing; Revision 2

2001-0022; Diesel Generator Heat Exchanger Service Water Flow Loop Uncertainty/Setpoint Calculation; Revision 2

2001-0049; Coordination-480V Switchgear; Revision 0

2002-0002; Nitrogen Backup System for MDAFP Discharge Valves (AF-4012/4019) and Minimum Flow Recirculation Valves (AF-4007/4014); Revision 3

2002-0003-00-C; Service water System Design Basis; dated July 25, 2003

2003-0008, CCW HX Plugging Limit; Revision 0

2003-0011; Determination of Relation Between SI Accumulator Leakage and SI Pump Discharge Pipe Venting Interval; Revision 3

2003-0014; MOV Operating Parameters; Revision 4

2003-0062; AFW Pump NPSH Calculation and Condensate Storage Tank Required Fluid to Prevent Vortexing; Revision 2

2004-0009; Safety Injection Pump Motor Protection; Revision 1

2004-0020; Estimated Containment Temperature resulting from an Appendix R Scenario; dated July 14, 2004

2004-0030; 480V MCC and Power Panel Coordination Calculation; Revision 0

2005-0002; AC Electrical System Analysis; Revision 0

2005-0008; Minimum Voltage Requirements for Safety Related Motor Control Center (MCC) Circuits, Revision 0

2005-0011; AFW Thermal Hydraulic Flow Model; Revision 1

2005-15; Motor Driven Auxiliary Feedwater Pump Motive Force; Revision 0

2005-0027; Aux Feedwater Flows During Main Steam line Break; Revision 0

2005-0028; Containment Air Temperature Indication Loop Uncertainty; Revision 0

2005-0031; Benchmarking of Training Simulator for Steam Generator Tube Rupture; Revision 0

Bechtel Engineering Calculations for Motor-Operated Valves for Wisconsin Electric Power Company; dated February 1988

CN-CRA-01-70; Point Beach SLB and Containment Response at 102% of 1524.5 Mwt with FRV Failure; Revision 0

CN-CRA-05-016; Point Beach Margin to Overfill; Revision 1

CN-TA-01-036; Analysis for LONF Transient; Revision 0

CN-TA-01-131; Loss of Normal Feedwater Analysis; Revision 0

EE 2001-0036; CC HX Testing and Acceptance Criteria; Revision 0

M-09334-419-VNDG-1; G01/G02 Room – Steady State Temperature Calculation; Revision 0

N-88-034; EDG Room Ventilation Test Evaluation; dated May 31, 1988

N-88-049; CCW Heat Exchanger Overall Heat Transfer with Seacure Tubing; Revision 2

N-89-019; Steam Generator Inventories During One Hour of Station Blackout; Revision 2

N-89-073; Diesel Generator Exhaust Stacks Column Loading (WT + TH + SSE); Revision 0

N-89-082; Support Load Qualification of Diesel Generator Piping Supports HB-29-H6, H7, H14, and H14; Revision 0

N-90-188; SI Pump Protective Relay Settings; October 25, 1990

N-91-016; PBNP Diesel Generator Loading Analysis; Revision 2

N-91-039; Safeguards Transformer Protection; dated April 2, 1991

N-93-057; Battery D06 DC System Sizing Voltage Drop and Short Circuit Calculations; Revision 5

N-93-058; Battery D05 DC System Sizing Voltage Drop and Short Circuit Calculations; Revision 5

N-93-058; Battery D105 DC System Sizing Voltage Drop and Short Circuit Calculations; Revision 4

N-93-059; Battery D106 DC System Sizing Voltage Drop and Short Circuit Calculations; Revision 4

N-93-82; SW-2816, SW-2817, 1(2)SW-2880 (Group 23) MOV Differential Pressure Calculation; Revision 0

N-93-86; 1(2)AF-4000, 4001 (Group 30) MOV Differential Pressure Calculation; Revision 0

N–94-059; CCW, HX-012A-D, Service Water Flow Verses Temperature Requirement; Revision 3

N-94-061; Minimum Usable Level in the RWST with Vortexing; Revision 1

N-94-064; VNBI (HX-105A/B) Service Water Flow vs. Temperature requirements; Revision 3

N-94-082; Service Water Flow Balance for Hot Shutdown After Appendix R Fires; Revision 2

N-94-110; PBNP Diesel Generator Addition – Transformer Safety Injection Pump Motor Protection; dated August 23, 1994

N-94-111; PBNP Diesel Generator Addition – Transformer 1X14(2X14) Protection; dated August 23, 1994

N-94-112; Diesel Generator Addition – Transformer 1X06 (2X06) Protection; Revision 1

N-94-124; PBNP Diesel Generator Addition – Bus Supply Protection and Coordination; Revision 1

N-94-130; 4160V and 480V Safeguards Buses Loss of Voltage Relay; Revision 1

97-0102; Containment Spray Duration for Use in Large Break LOCA Dose Calculations; dated May 14, 1997

N-97-0135; Diesel Generator G01/G02 Bus Supply Coordination; dated June 24, 1997

N-98-095; Minimum DC Control Voltage; Revision 01-A

NRC-02-070; Resolution of Generic Letter 96-06 Waterhammer Issues; dated July 30, 2002

NSD-SAE-ESI-99-074; Steam Line Break Mass and Energy Release and Containment Analysis; dated February 24, 1999

P-89-031; Voltage Drop Across MOV Power Lines; Revision 10

P-90-017; Motor Operated Valve Undervoltage Stem Thrust and Torque Calculation; Revision 20

P-94-002; Condensate Stroage Tank (t-24A/B) Level Alarm Heights; Revision 0

P-94-005; MOV Stem Thrust Calculation for Gate and Globe Valves; Revision 10

PBNIC-34; Refuel Water Storage Tank Level Scaling; Revision 0

PBNP-IC-42; Condensate Storage Tank Water Level Instrument Loop Uncertainty/Setpoint Calculation; Revision 0

PGT-2002-1189; Component Cooling Water Heat Exchangers HX-012A and HX-012B Thermal Performance Test Data Evaluation and Uncertainty Analysis; Revision 0

PGT-2002-1270; Component Cooling Water Heat Exchangers HX-012C and HX-012D Thermal Performance Test Data Evaluation and Uncertainty Analysis; Revision 0

PI-PB-13; Design Control Summary Design Verification for Pipe Supports; Revision 5

Report No. 457641; Valve Data and Thrust Calculations; dated July 11, 1991

WE-100145; Unit 1 RWST IT13 from valve 1SF-811; Revision 0

WE-100145-00-A; RWST 1T13 from valve 1SF-811; Revision 0

WE-100150; SFP Recirc Piping to Unit 1 RHR; Revision 0

WE-100150-00-A; SFP Recirc Piping to Unit 1 RHR; Revision 0

WE-100150-00-B; SFP Recirc Piping to Unit 1 RHR; Revision 0

WE-200130; RWST Piping from valve 2SF-819A to header 10 IN.-AC-601R-2; Revision 0

WE-200130-00-A; RWST Piping from valve 2SF-819A to header 10 IN.-AC-601R-2; Revision 0

WE-200131; RWST Piping from valve 2SF-811 to tank 2T-13; Revision 0

WE-200131-00-A; RWST Piping from valve 2SF-819A to header 10 IN.-AC-601R-2; Revision 0

WE-200131-00-B; RWST Piping from valve 2SF-811 to tank 2T-13; Revision 0

WE-200131-00-C; RWST Piping from valve 2SF-811 to tank 2T-13; Revision 0

WE-300054; Unit 1 and 2 RWST Spent Fuel Pit Demineralizer Piping; Revision 0

WE-300054-00-A; Unit 1 and 2 RWST Spent Fuel Pit Demineralizer Piping; Revision 0

WE-300054-00-B; Unit 1 and 2 RWST Spent Fuel Pit Demineralizer Piping; Revision 0

WE-300054-00-C; Unit 1 and 2 RWST Spent Fuel Pit Demineralizer Piping; Revision 0

WE-300064; 2 IN. -AC-151R-3/5, SFP Recirc Piping; Revision 0

WE-300064-00-A; SFP Recirc Piping - DI and P-13 Discharge Branches; Revision 0

WE-300066; P-13 SFP Skimmer Pump Suction Piping; Revision 0

WE-300066-00-A; P-13 SFP Skimmer Pump Suction Piping; Revision 0

WE-300067; Small Bore Piping Support Qualification; Revision 0

WE-300067-00-A; RWST Small Bore Piping Support Qualification; Revision 0

<u>Corrective Action Documents Generated Due to the Inspection</u>

AR 01044417; Tech Spec Basis B 3.6.6 (Spray and Cooling systems) incorrectly asserts LOCA is limiting transient for containment pressure and states Steam line break (SLB) is less limiting; dated August 30, 2006

AR 01044583; Revise OM 4.3.2 to remove AFW isolation action assumption; dated August 31, 2006

AR 01044646; OM 4.3.2, page 52 stated that operator timing was to be done in 2004, but changes were not submitted; dated August 30, 2006

AR 01044692; 50.59 process applied but not used for a 2005 Tech. Spec Bases change; dated August 30, 2006

AR 01044780 ; Leakage value attributed to 1SI-829C, HHSI High Flow Test Line Manual Flow Control Valve (FCV) was in error; dated August 30, 2006

AR 01044972 ; P.A. model enhancement identified; dated August 31, 2006

AR 01045064; Discontinuity between PRA model and the human error probability (HEP) documentation was discovered; dated August 30, 2006.

AR 01045086; P.A. model enhancement identified; dated August 31, 2006

AR 01045089; P.A. model enhancement identified; dated August 31, 2006

AR 01045094; P.A. model enhancement identified; dated August 31, 2006

AR 01045326; Closeout unjustified for 6 OE items; dated August 30, 2006

AR 01045713; Over-conservative input data for 125 VDC calculations; date September 7, 2006

AR 01046002; Fire Water to TDAFPs not checked for degradation; dated August 30, 2006

AR 01046916; Typographical error found in 10CFR 50.59/72.48 screening 2001-0281, MR 00-093 Contractor upgrade and Wiring improvements for RWST Immersion Heater; dated August 31, 2006

AR 01046949; Incorrect wording in 50.59 screening for TM 02-0003; dated August 30, 2006

AR 01046952; TS Basis 3.8.1 Revision 0 in EDMS not Rev 0; dated August 30, 2006

AR 01047051; ECA-1.3, Unit 1 not scanned properly in EDMS or Sharepoint; dated August 30, 2006

AR 01047170; Evaluate AOPs and EOP set for use of “Cautions and Notes”; dated August 30, 2006

AR 01047207; It was noted that the licensing basis information in DBD-16 needs to be updated; dated August 30, 2006

AR 01047154; SP2168.15 Revision 7 POST LOCA HYDROGEN CONTROL SYSTEM PRESSURE. Specify gauge calibration and range requirements; dated August 31, 2006

AR 01047162; ECA-1.3, Unit 2. Evaluate and revise ECA-1.3 to include the SI pump minimum flow value prior to step 12; dated August 31, 2006

AR 01047265; Potential Inadequate Implementation - BUL 88-04. ECA-1.3 (both units) did not include SI pump minimum flow value prior to Step 12; dated August 31, 2006

AR 01047351; Boron concentrations used in MSLB analysis; August 30, 2006

AR 01047353; A review of OPR000153 was performed and identified that effects of a seismic event were not considered in OPR; dated August 31, 2006

AR 01047363; Bases for TS 3.5.1 is not correct (Boron concentration); dated August 30, 2006

AR 01047372; Inadequate administrative limits on RWST and SI accumulators boron concentrations; dated September 7, 2006

AR 01047394; During a JPM walkthrough it was discovered local plaque was missing from 1 SI-897A which states that valve is “counter-clockwise to close”; dated September 7, 2006

AR 01047496; CFC Thermal Performance Testing Issues and Plan (GL 89-13); dated September 7, 2006

AR 01047526; While it appears that PBNP uses WCAP 15603 Revision 1-A; however, it is not clear that station has explicitly met them by evaluation or other documentation; dated August 31, 2006

AR 01047564; Revise SEP-2.1 to include caution concerning minimum flow requirements for RHR/SI Pumps; dated August 31, 2006

AR 01047568; Revise SEP-2.1 to include caution concerning minimum flow requirements for RHR/SI Pumps; dated August 31, 2006

AR 01047573; Revise SEP-2.3 to include caution concerning minimum flow requirements for RHR/SI Pumps; dated August 31, 2006

AR 01047577; Revise SEP-2.3 to include caution concerning minimum flow requirements for RHR/SI Pumps; dated August 31, 2006

AR 01047592; FSAR May Omit Some Tornado License Bases; dated August 31, 2006

AR 01047610; Tornado Missile Design and License Basis Unclear; dated August 31, 2006

AR 01047624; BG SEP-2.1 Update to add caution for RHR and SI minimum Flows; dated August 31, 2006

AR 01047631; BG SEP-2.3 to include RHR/SI minimum flows; dated August 31, 2006

AR 01047643; BG ECA-1.3 Update BG ECA-1.3 with SI pump minimum flow value; dated August 31, 2006

AR 01047645; DD ECA-1.3 Update DD ECA 1.3 to include SI pump minimum flow value; dated August 31, 2006

AR 01047819; Lack of formal qualification for CC pipe support MR 96-077 modified pipe support AC-152N-12-H203; dated August 31, 2006

AR 01047821; Vortexing not considered for CST levels during SBO; dated September 7, 2006

AR 01047828; Incorrect operability call for AR00408837. Calculation 95-0102 and design guide DG-E09 are non-conservative for fire wrap derating for the purpose of cable ampacity; dated August 31, 2006

AR 01047912; EDG frequency not accounted for in static calculation; dated September 1, 2006

AR 01048072; Verification of 10 minute timing to align EDG to a blackout unit; dated September 7, 2006

AR 01048547; EDG frequency hydraulic analysis concerns; dated September 7, 2006

AR 01048558; Inappropriate change to EOP-1.3 involving containment spray times; dated September 7, 2006

AR 01048599; Containment Spray duration not consistent with calc assumption; dated September 7, 2006

AR 01048857; Inconsistencies found in FSAR LOCA analyses; dated September 7, 2006

AR 01049147; Non-retrievable 50.59 documentation; dated September 11, 2006

AR 01049163; License basis for containment penetration 19 is not clear; dated September 11, 2006

AR 01049307; CST level/volume calculation; dated September 11, 2006

AR 01049322; OPRs not being reviewed as associated documents; dated September 11, 2006

AR 01049585; Incorrect assumption in Calc –91-016 that charging pump must be started; dated September 12, 2006

AR 01049620; SW pump oil not identified in fire hazards report; dated September 12, 2006

AR01049647; Missed loads in EDG static loading calculation; dated September 12, 2006

AR 01049659; Typo in calc 2005-0011, Revision 1; dated September 12, 2006

AR 01049672; Testing concern for AF-133; dated September 13, 2006

AR 01049700; Error on Bechtel drawing P-442, Sh. 10, piping support drawing; dated September 12, 2006

AR 01050110; Inadequate change to commitments in GL 89-13; September 14, 2006

AR 01050127; Admin step 6.1.2 not signed off in completed test procedure; dated September 14, 2006

AR 01050174; EDG Control Cubicle rear door not fully secured; dated September 14, 2006

AR 01050201; Notification commitment for IEB 88-04 not met; dated September 14, 2006

AR01050213; P&ID drawing deficiency related to isolation valves for flow transmitter 1-FE-925; dated September 14, 2006

AR 01050348; Unable to locate 50.59 for 1991 FSAR change; dated September 15, 2006

AR 01050361; Non-conservatisms in the EDG Room Heatup Calculations; dated September 15, 2006

AR 01050377; Scaffold not evaluated by engineering; dated September 15, 2006

AR 01050458; Inconsistencies in operations procedures in regard to locking valve manual handwheel; dated September 15, 2006

AR 01050473; Inadequate/ineffective compensatory measures for bolted fault conditions; dated September 15, 2006

AR 01050637; Error in qualification of pipe support AC 151R-4-SN2; dated September 21,2006

AR 01050640; Error in qualification of pipe support AC 151R-4-SN3; dated September 21, 2006

AR 01051042; Controls needed to keep EDG rubber flaps clear of snow/ice; dated September 19, 2006

AR 01051112; GL 89-13 Program and Commitment changes not in FSAR; dated September 19, 2006

AR 01051182; Cables found outside cable trays; dated September 27, 2006

AR 01051297; Typographical errors in AOP-9B; dated September 20, 2006

AR01051488;TDAFP Room Heatup Calculation does not exist for 4 hour coping duration; dated September 21, 2006

AR 01051496; CFC Factor low limit of 0.0003 is not demonstrated; dated September 27, 2006

AR 01051574; OPR 153 did not address effects of non-safety related cables routed in raceways with safety related cables; dated September 21, 2006

AR 01051661; EOPs contain actions that hinder sump recirculation; dated September 22, 2006

AR 01051786; EDG Day Tank calculation missing fireproofing loads; dated September 25, 2006

AR 01051821; EVAC Treatment Effect on Diesel Coolers; dated September 23, 2006

AR 01051831; Maximum dP for MOVs 1 and 2 AF-4000 and 4001 in calculation–93-86;dated September 24, 2006

AR 01052014; WE-300066 does not consider local stresses; dated September 25, 2006

AR 01052043; No basis for operability deflection in WE-300066; dated September 25, 2006

AR 01052205; Technical Specification Basis for 3.7.7 contains incorrect statement concerning isolation capability of CCW subsystem for accident mitigation; dated September 26, 2006

AR 01052214; Upper limit temperature for CST should be changed to 95 degrees; dated September 26, 2006

AR 01052220; Upper limit temperature for CST should be changed to 95 degrees in operator rounds; dated September 26, 2006

AR 01052281; Cable found outside of Cable Tray CL01; dated September 27, 2006

AR 01052312; No AFW Pump Room Heatup Calculation to support 4 hour SBO coping duration; dated September 27, 2006

AR 01052415; Justification of frequency effects on hydraulic analysis; dated September 27, 2006

AR 01052446; Calculation WE-300054 contains deficiencies in relation to Pipe Support S8; dated September 27, 2006

AR 01052513; Reduced moments applied inappropriately in structural calculation WE-300067; dated September 28, 2006

AR 01052554; Loose anchor bolts justified inappropriately in calculation; dated September 28, 2006

AR 01052649; CDBI-Piping analysis calculation requires clarification; dated September 28, 2006

AR 01053234; Assumption in Calc –94-059 for CCW flow does not reflect most current data; dated October 2, 2006

AR 01053357; Potential issue associated with adequacy of EDG Surveillance Requirement to meet TS 3.8.1; dated October 4, 2006

AR 01053632; During the NRC CDBI, greater than 80 CAPs were generated; dated October 4, 2006

<u>Corrective Action Documents Reviewed During the Inspection</u>

ACE001923; Failure of Degraded Voltage Relay to Trip During 1RMP-9056; dated August 22, 2005

AR 00005496; DBDOI Tracking for DBD-T-36; dated November 27, 2001

AR 00400406; Non-Compliance With FSAR for Cable Overload Protection; dated January 22, 2003

AR 00464451; Cable Ampacity Concern; dated June 1, 2003

AR 00524983; 125 VDC System Assessment; dated September 6, 2003

AR 00567895; Calculation N-92-004 Not Being Updated for Breaker Changes; dated August 20, 2004

AR 00575756; Various Calculation Items Identified; dated August 20, 2004

AR 00600384; IT-10 Acceptance Criteria Does Not Ensure Adequate AFW Flow Without Operator Action; dated April 28, 2004

AR 00752940; Failure to Include EDG Frequency variation in Hydraulic Analyses; dated November 11, 2004

AR 00772885; Operator Action to Not Overfill Ruptured SG; dated November 4, 2004

AR 00806888; Minor Deficiencies in Ventilation calculation; dated February 15, 2005

AR 00825876; Review of KNPP Emergency Diesel Exhaust Duct Concern; dated March 29, 2005

AR 00889394; Calculated Short Circuit Currents Exceed Equipment Rating; dated September 22, 2005

AR 00889745; Overload Concerns of Safety Related Equipment; dated September 22, 2005

AR 0889747; Non-Conservative Tech Spec and Degraded Voltage Time Delay Settings; dated September 22, 2005

AR 00901285; Simulator Modeling of SGTR Break Flow Not Accurate (Non-Conservative); dated November 21, 2005

AR 00909252; Cumulative Risk Impact of HX-12A Being Unavailable; dated March 8, 2006

AR 01018818; Request an OE Evaluation of OE22166 - Air Void in ECCS Recirculation Piping at Palo Verde; dated March 16, 2006

AR 01023171; Operability Recommendation; The Main Steam Safety Valves (MSSVs); Revision 0

CAP 028946; SSDI Question No. 43, EOP 1.3 Manual CC Valves; dated August 5, 2002

CAP 0404661; Analysis for AFW Pumps DP is Non Conservative for IST Test Criteria; dated February 5, 2003

CAP 050340; Determine Safety Function of Component Cooling Water System Manual Valves; dated September 22, 2003

CAP 051581; VNPSE valves IST acceptance criteria not conservative; dated November 3, 2003

CAP 052962; Issues Raised During Review of Component Cooling Water System Procedure Review; dated January 19, 2004

CAP057853; Non Conservative Service Water System Pressures Used in MOV Analysis; dated July 9, 2004

CAP 059243; Failure to Include EDG Frequency Variation in Hydraulic Analyses; dated September 17, 2004

CAP 0600384; IT-10 Acceptance Criteria Does Not Ensure Adequate AFW Without operator Action; dated June 4, 2004

CAP 061981; Ventilation Calculation Review Finds Minor Deficiencies; dated February 10, 2005

OPR000044; MSSV Setpoint Tolerance Was Not Included in MDAFW Pump IST Acceptance Criteria Calculation; Revision 1

OPR 000109; Potential Variability of the Emergency Diesel Generator Provided Frequency During a Loss of Offsite Power Event; dated September 17, 2004

OPR000109; IT-10 Acceptance Criteria Does Not Ensure Adequate AFW Without Operator Action; Revision 1

OPR 112; G-01, 3 and 4; Revision 1

OPR000120; Postulated Debris in Fire Water May Challenge TDAFP Operability; dated January 31, 2005

OPR 000148; Potential to Crimp AFW Pump Recirculation Lines; Revision 1

OPR 153; Calculated Short Circuit Currents Exceed Equipment Rating; dated September 22, 2005

OPR 154; Overload Concerns of Safety Related Equipment; Revision 1

OPR 155; 1X-13 Station Service Transformer, 2X-14 Station Service Transformer; Revision 0

OPR 156; Non-Conservative Tech Spec and Degraded Voltage Time Delay Settings;

Revision 0

OTH014067; Evaluate Enhancing the Flow Check of FW Supply to TDAFW Pumps; dated July 15, 2004

RCE 000044; U2 Safety Injection Pump “Gas Bound” During Routine Preventive Maintenance; Revision 1

Design Changes/Modifications

88-099; Modify the AFW Pump Mini-Recirc. Lines to Provide the Recommended Flow rates Replacement Transmitter for 2LT-935 and 2LT-939 Accumulator Level Transmitter; dated February 28, 2004

94-066*A; Soft face Check Valve Disk (CSI-834D) and relief valve Installation (include design packages 94-066*A-01 and 94-066*A-02); dated August 5, 1996

95-048; U1 EDG Output Breaker Undervoltage Permissive Time Delay; dated April 6, 2001

98-021; U0 Small Bore Pipe Support Upgrades for RWST and SFP Piping; dated 1999

01-098; Upgrade Service Water Zurn Strainer D/P Indication and Alarm Instrumentation; dated November 28, 2001

02-044; Installation of Temporary Relief Valve for SI-830B, T-34B SI Accumulator Relief; dated June 30, 2000

TM 04-001; dated Temporary Replacement of Unit 1 Purge Supply/Return Valves; dated June 2004

Drawings

346307; Elementary Wiring Diagram Auxiliary Feedwater Pump P-38B Point Beach N.P. Unit 2; Revision 01

6118-C-181; Concrete Turb. Building – Class 1 Structure Plans at El. 26’-0 and El. 44’-0; Revision 15

6118-E-27; Raceway System Notes and Details; Revision 2

EAFK 141002; Logic Diagram Auxiliary Feedwater Discharge MOV Control Logic; Revision 01

EAFS 000001; Elementary Wiring Diagram Turbine Driven Auxiliary Feedwater Pump Mini Recirc Control Valve 1AF-4002; Revision 15

EAFS 000002; Elementary Wiring Diagram Turbine Driven Auxiliary Feedwater Pump Mini Recirc Control Valve 2AF-4002 Point Beach N.P. Unit 2; Revision 12

EAFS 165001; Elementary Wiring Diagram P38A Auxiliary Feedwater Pumps Suction From Service Water AF-4009 Point Beach N.P. Unit 1; Revision 07

EAFS 165002; Elementary Wiring Diagram Turbine Driven Auxiliary Feedwater Trip/Throttle Valve 1MS-02082 Point Beach N.P. Unit 1; Revision 02

EAFS 165002; Elementary Wiring Diagram Turbine Driven Auxiliary Feedwater Trip Throttle Valve 2MS-02082 Point Beach N.P. Unit 2; Revision 04

EAFS 165003; Elementary Wiring Diagram P38B Auxiliary Feedwater Pump Discharge to 1HX-1B Steam Generator AF-4021 Point Beach N.P. Unit 1and2; Revision 07

EAFS 165006; Elementary Wiring Diagram P38A Auxiliary Feedwater Pump Discharge to 1HX-1A Steam Generator AF-4023 Point Beach N.P. Unit 1and2; Revision 08

EAPS 000002; Stm. to Turb. Dr. AFWP 2P29 2MOV2019 [2MOV2020] Low Suction Press. Ckt. Point Beach N.P. Unit 2; Revision 06

EAPS 000007; Stm. to Turb. AFWP 1P29 1MOV2019 [1MOV2020] Low Suction Pess. Ckt. Point Beach N.P. Unit 1; Revision 10

EAPS 000067; Elementary Wiring Diagram 1P-29 Turbine Driven AFP Steam Supply MOV 1MS-2020; Revision 02

EAPS 000068; Elementary Wiring Diagram 1P-29 Turbine Driven AFP Steam Supply MOV 1MS-2019; Revision 03

EAPS 000096; Elementary Wiring Diag. Steam to Turb Dr Aux F.P. 2MOV2019 [2MOV2020] Point Beach N.P. Unit 2; Revision 21

EAPS 240001; Meter and Relay Diagram 4160 V Auxiliary System; Revision 07

EAPS 240009; Elementary Wiring Diagram 4160V Switchgear Bus 1A05 Undervoltage and Diff. L.O. Relays; Revision 07

EAPS 240044 Sh. 1; Schematic Diagram 4160V SWGR Bus 1-A06 (2-A06) Undervoltage and Diff. L.O. Relay Schemes, Revision 16

EAPW 128001; Three Line Relay and Metering EDG G-03 4160V Bus 1-A06; Revision 09

EFWS 165007; Elementary Wiring Diagram Auxiliary Feedwater Pump P-38 Automatic Actuation Point Beach N.P. Unit 2; Revision 02

EFWS 165008; Elementary Wiring Diagram Auxiliary Feedwater Pump P-38 Automatic Actuation Point Beach N.P. Unit 2; Revision 01

EFWS 165009; Elementary Wiring Diagram Auxiliary Feed Pump P-38A Control Circuit for Feeder Breaker Point Beach N.P. Unit 1; Revision 03

EFWS 165009; Elementary Wiring Diagram Power to P-038B-M Aux Feedwater Motor Driven Pump Point Beach N.P. Unit 2, Revision 01

EFWS 165010; Elementary Wiring Diagram Auxiliary Feed Pump P-38B Control Circuit for Feeder Breaker Point Beach N.P. Unit 2; Revision 03

EPLS 141001; Miscellaneous Relay Elementary Diagram 2C-158 Point Beach N.P. Unit 1; Revision 03

EPLS 141005; Miscellaneous Relay Elementary Diagram 1C-158 Point Beach N.P. Unit 1; Revision 04

ERPK 141002; Point Beach Nuclear Plant Units 1 and 2 Logic Diagrams Safeguards Actuation Signal; Revision 21

ERPK 141006; Logical Diagram Stema Generator Trip Signals Point Beach N.P. Units 1 and 2; Revision 09

ERPL 000001; Logic Diagram Safeguard Sequence Point Beach N.P. Units 1 and 2; Revision 18

ERPS 141078; Schematic Diagram, SI Logic Engineered Safety Features (ESF) Systems Train “A” Reactor Safeguards Systems Point Beach N.P. Unit 1; Revision 19

ERPS 141079; Schematic Diagram, SI Logic Engineered Safety Features (ESF) Systems Train “B” Reactor Safeguards Systems Point Beach N.P. Unit 1; Revision 18

ETGS 141001;Schematic Diagram Gas Turbine Control DC Control Point Beach N.P.; Revision 15

EWFS 000003; Schematic Diagram Auxiliary Feedwater Pumps Control Point Beach Nuclear Plant; Revision 14

FFWS 000005; Elementary Wiring Diagram 2P-29 Turbine Driven Aux. Feedwater Pump Start on Bus 2A01 and 2A02 Undervoltage; Revision 03

M-207, Sheet 1; Service Water; Revision 68

M-207, Sheet 1A; Service Water; Revision 26

M-209 Sheet 3; Instrument Air; Revision 13

M-217, Sheet 1; Auxiliary Feedwater System; Revision 82

M-217, Sheet 2; Auxiliary Feedwater System; Revision 22

P-102; Piping Isometric Safety Injection V-827A and B / T-13 to SI Pump P-15A and B, P-14A and Band V-856A and B SI-151R; Revision 12

P-117; Aux Feedwater Pump Suction from Condensate Storage Tanks; Revision 11

P-118; Aux. F.W. Pump Suction from Condensate Storage Tanks 1-T24A and B; Revision 7

P-119; Piping Isometric Safety Injection from Pump P-15A and B to Containment Penetration P-27 and P-13 4" and 6" SI-1501R-1, -2, and -3; Revision 11

P-120; Piping Isometric Hot Leg Injection Line from Penetration P27 to Reactor Coolant System 4-SI-1501R-2 Unit 1; Revision 7

P-131; Piping Isometric Cold Leg Injection from Penetration P13 to Reactor Coolant System 4-SI-1501R-3; Revision 6

P-132; Piping Isometric Auxiliary Coolant System to RHR Pumps P10A and B Suction and Discharge Outside Containment Unit 1; Revision 10

P-133 Sheet 1; Piping Isometric RHR Heat Exchanger Discharge and Bypass AC-501R-S-6 Outside Containment Unit 1; Revision 14

P-133 Sheet 2; Piping Isometric RHR Test Line AC-601R-7; Revision 3

P-134; Piping Isometric RHR Heat Exchanger Discharge and Bypass AC-601R-6 Unit 1; Revision 8

P-136; Piping Isometric SIS from Penetration P-22 to Reactor Vessel 1-R1 SI-601R-2, SI-2501R-5, RC-2501R-5 Unit 1; Revision 8

P-137; Piping Isometric SIS from Valve 841 to Primary Coolant Cold Leg 6"-SI-2501R-1, 2, 4 10"-SI-2501R-1, 2 10"-RC-2501R-5, 8; Revision 10

PB 02 MSIK00000150; P&ID Safety Injection System; Revision 38

PBA-1140 Sheet 1; Piping Isometric 2" Safety Injection (SI) Piping in Containment Unit 1; Revision 1

PBA-1140 Sheet 2; Piping Isometric 2" Safety Injection (SI) Piping in Containment Unit 1; Revision 1

PBA-1140 Sheet 3; Piping Isometric 2" Safety Injection (SI) Piping in Containment Unit 1; Revision 2

P&ID WE PBM-230; Radwaste Component Cooling Water; Revision 13

P&ID WEST 110E017 Sheet 1; Safety Injection System Unit 1; Revision 53

P&ID WEST 110E017 Sheet 2; Safety Injection System Unit 1; Revision 55

P&ID WEST 110E017 Sheet 3; Safety Injection System Unit 1; Revision 46

P&ID WEST 110E018 Sheet 1; Auxiliary Coolant System Unit 1; Revision 57

P&ID WEST 110E018 Sheet 2; Auxiliary Coolant System Unit 1; Revision 21

P&ID WEST 110E018 Sheet 3; Auxiliary Coolant System Unit 1; Revision 41

Miscellaneous Documents

05-1; Specification for Electrical Installation for Modifications to Point Beach Nuclear Plant - Unit 1 and 2; dated April 17, 1981

10 CFR 21-0082; ESI Part 21 Report for Woodward Electronic Controls with Electrolytic Capacitors; Revision 2

BC-TOP; Design of Structures for Missile Impact; Revision 2

B-TOP-3; Design Criteria for Nuclear Power Plants Against Tornadoes; dated March 12, 1970

Byron Jackson Letter to WE; Byron Jackson Job No. 891-C-2264.21, Minimum Flow Analysis; dated August 7, 1989

DBD-019; 125 VDC System; Revision 7

DG-C01; Wisconsin Electric Guideline for Design, Qualification, and Installation of Concrete Expansion Anchors AT PBNP; Revision 0

DG-C03; Seismic Design Criteria Guideline; Revision 0

DG-E-07; Separation of Electrical Cables; Revision 2

DG-E09; Cable Ampacity Calculations; Revision 1

DG-E13; Insulated Electrical Cable Installation; Revision 0

DG-M10; Pipe Support Guidelines; Revision 2

Engineering Evaluation 2001-0036; CC Heat Exchanger Testing and Acceptance Criteria; dated November 27, 2001

Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Power Plant Equipment; Revision 2

FR-H.1; Response to Loss of Secondary Heat Sink; Revision 1C

Heat Exchanger Specification sheet, Component Cooling Heat Exchanger; dated February 24, 1992

HI-2002418; Holtec report No. HI-2002418, Thermal Performance of Containment Fan Coolers; Revision 1

HX-01; Heat Exchanger Condition Assessment Program; Revision 2

Information Exchange on Electrical Design; dated September 1985

LER 97-021; Spent Fuel Pool Cooling System Not In Accordance With Plant Design Basis; Revision 0

Letter, VPNPD088-335, NRC-88-062; Potential Safety Related Pump Loss; dated June 28, 1988

Letter from Byron Jackson to WE; Safety Injection Pump Minimum Flow Capacity Pump S/N 671-N-0446/9; dated March 20, 1988

Letter from Wisconsin Electric to NRC; IE Bulletin 80-04, Analysis of PWR Main Steam Line Break with Continued Feedwater Addition; dated April 25, 1980

Letter from Wisconsin Electric to NRC; IE Bulletin 80-04, Analysis of PWR Main Steam Line Break with Continued Feedwater Addition; dated April 14, 1982

Letter from Wisconsin Electric to NRC; IE Bulletin 80-04, Analysis of PWR Main Steam Line Break with Continued Feedwater Addition; dated March 23, 1988

Letter from Wisconsin Electric to NRC; IE Bulletin 80-04, Analysis of PWR Main Steam Line Break with Continued Feedwater Addition; dated September 7, 1988

Letter from Wisconsin Electric to NRC; Supplement to Technical Specifications Change Request 192; dated June 3, 1997

Letter, NEL-90-93; EMD Emergency Diesel Generators; dated March 26, 1990

Letter, Morrison-Knudsen Company, Inc.; Wisconsin Electric Letter NEL-90-93; dated May 15, 1990

Letter, Wisconsin Electric to NRC; NRC Bulletin 88-04 Potential Safety-Related Pump Loss; dated June 28, 1988

Letter, NRC to Wisconsin Electric; Response to NRC Bulletin 88-04; dated May 26, 1989

Letter, Flowserve to Wisconsin Electric; Aux. Feed Water Pumps Minimum Flow Analysis; dated March 21, 2001

Letter, Wisconsin Electric to NRC; Clarification of a Commitment Concerning Operation of the Safety Injection and Residual Heat Removal Pumps; dated February 26, 1993

NDC-27204; Flooding Resulting From Non-Category I Failure; dated February 17, 1975

NPL 2000-0451; Reply to a Request for Additional Information to Generic Letter 96-06; dated October 12, 2000

NPM 1992-0428; Memo from A.R. Bayer to Holders of Design and Installation Guidelines Manual, “Design Guideline DG-C01 Revisions”; dated April 27, 1992

NPM 2003-0330; Evaluation RIS 01-015 “DC MOV Capability and Stroke Time”; dated May 12, 2003

NRC-89-043; Response to 10 CFR 50.63, Loss of all Alternating Current Power; dated April 17, 1989

NRC-89-114; Supplement to 10 CFR 50.63, Loss of all Alternating Current Power; dated September 26, 1989

NRC-90-030; Supplement to 10 CFR 50.63, Loss of all Alternating Current Power; dated March 30,1990

NRC-91-134; Technical Specification Change Request 144 Condensate Storage Tank Level Requirements; dated April 24, 1991

NRC-93-061; Supplement to 10 CFR 50.63, Loss of all Alternating Current Power; dated May 14, 1993

NRC-93-101; Design Summary for the Installation of Two Additional Emergency Diesel Generators; dated September 24, 1993

NRC-94-041; Technical Specification Change Request 166; May 26, 1994

NRC 2002-022; License Amendment Request 225, Technical Specification LCO 3.7.8, Service Water; dated March 20, 2002

NRC SER; Safety Evaluation of the Point Beach Response to the Station Blackout Rule; dated October 3, 1990

NRC SER; Amendment Mos. 152 and 156; dated September 23, 1994

NRC SER; Issuance of Amendments Re: Technical Specification 3.9.3, Containment Penetrations, Associated with Handling of Irradiated Fuel Assemblies and Use of Selective Implementation of the Alternative Source Term for Fuel Handling Accident; dated April 2, 2004

PB-546; Specification for Electrical Installation; Revision 1

Probabilistic Safety Assessment High Winds and Others Notebook Section 9; dated July,1995

PBSA-ENG-06-02; Focused Self-Assessment Preparation for Design Basis Inspection Based on 71111.21; dated January 16, 2006 through February 3, 2006

Rep-0774-01; USNRC Generic Letter 87-02 Unresolved Safety Issue A-46 Resolution Seismic Evaluation Report; Revision 1

SCR 2002-0247; Transfer to Containment Sump Recirculation; dated June ???

SCR 2002-0295; Changes to EOP-1 and EOP-1.3 to Eliminate the Requirement for the SI Core Deluge Flowpath; dated August 1, 2002

SCR 2002-0306; Transfer to Containment Sump Recirculation; dated August 6, 2002

SCR 2003-0394; TM 03-036, Install Blank Flange at 2VNPSE-3212 and 2VNPSE-3244; dated November 8, 2003

SCR 2004-0180; Uni1 and Unit 2 ECA-0.0, Loss of All AC Power, Revision 37 for Unit 1/Revision 38 for Unit 2; dated July 19, 2004

SE 2001-007; Component Cooling Water System Closed Loop Inside Containment; dated February 24, 2001

Structural Design Criteria for the Point Beach Nuclear Plant, by Bechtel Corporation, July 1967; Revision 2

T-29986-2; SI Pump Curve; dated October 3, 1968

TC-10443; Service Water Pump Curve; Revision 0

TLB 34; Condensate Storage Tank T24A/B); Revision 34

WE letter to NRC, Potential Safety Related Pump Loss; dated June 28, 1988

WEP-01-060; Containment Response to Steamline Break at 1524.5 Mwt NSSS Power - Final Report; dated October 29, 2001

Procedures

00367A; EMD Engine Derating for Elevated temperatures; dated June 9, 1992

2-CL-CC-001; Component Cooling Unit 2; Revision 10

AOP-5B; Loss of Instrument Air; Revision 28

AOP-9B Unit 1; Component Cooling System Malfunction; Revision 19

AOP-23 Unit 1; Establishing Alternate AFW Suction Supply; Revision 6

ARB 2C20 A 4-2; White/Yellow Bus Ventilation Trouble; Revision 4

BG-CSP-H.1; Response to Loss of Secondary; Revision 23

CMP 2.2.7; Engineering Instructions for Performing Valve Operator Checkouts; Revisions 6, 10

CSP-H.1; Unit 1 RED, Response to Loss of Secondary Heat Sink; Revision 26

ECA-0.0 Unit 1; Loss of All AC Power; Revision 41

ECA-1.1 Unit 1; Loss of Containment Sump Recirculation; Revision 31

ECA-1.3; Containment Sump Blockage; Revision 1

EOP-0 Unit 1; Reactor Trip or Safety Injection; Revision 43

EOP-0 Unit 2; Reactor Trip or Safety Injection; Revision 44

EOP-0.1 Unit 1; Reactor Trip Response; Revision 31

EOP-1 Unit 1; Loss of Reactor or Secondary Coolant; Revision 38

EOP-1.2 Unit 1; Small Break LOCA Cooldown and Depressurization; Revision 27

EOP-1.3 Unit 1; Transfer to Containment Sump Recirculation - Low Head Injection; Revision 36

EOP-1.3 Unit 1; Transfer to Containment Sump Recirculation - Low Head Injection; Revision 36

EOP-1.4 Unit 1; Transfer to Containment Sump Recirculation - High Head Injection; Revision 17

EOP-3 Unit 1; Steam Generator Tube Rupture; Revision 37

ICP 06.006; Service Water System Non-Outage Instruments Calibration; November 23, 2004; Calibration date September 16, 2005

IT 01; High Head Safety Injection Pumps and Valves, Unit 1;dated November 28, 2005; Test date July 18, 2006

IT 02; High Head Safety Injection Pumps and Valves (Quarterly) Unit 2; dated November 28, 2005; Test date August 2, 2006

IT 03; Low Head Safety Injection Pumps and Valves (Quarterly) Unit 1; Revision 52

IT-07A; P-32 A Service water Pump (Quarterly); December 8, 2005, Test date August 9, 2006

IT 7F; P-32F Service Water Pump (Q); December 8, 2005; Test date May 31, 2006

IT 250; Chemical and Volume Control and Component Cooling System Valves (Cold Shutdown) Unit 1; Revision 24

IT 255; Chemical and Volume Control and Component Cooling System Valves (Cold Shutdown) Unit 2; Revision 21

IT 530A; Leakage Reduction and Preventive Maintenance Program Seat Leakage Test of the Train A RHR System, (Refueling) Unit 1; Revision 15

IT 530B; Leakage Reduction and Preventive Maintenance Program Seat Leakage Test of the Train B RHR System, (Refueling) Unit 1; Revision 15

IT 530C; Leakage Reduction and Preventive Maintenance Program Train “A” HHSI and RHR “Piggyback” Test (Refueling) Unit 1; Revision 11

IT 530D; Leakage Reduction and Preventive Maintenance Program Train “B” HHSI and RHR “Piggyback” Test (Refueling) Unit 1; Revision 13

IT 605; Radwaste Component Cooling Water Supply and Return Valves (Cold Shutdown) Unit 2; Revision 8

IT 800; Component Cooling Water System Valves (Bi-Annual) Unit 1; Revision 2

IT 805; Component Cooling Water System Valves (Bi-Annual) Unit 2; Revision 1

MI 32.9; Scaffolding Program; Revision 19

NP 1.9.6; Plant Cleanliness and Storage; Revision 18

NP 3.4.8; Requirements for Scaffold Nera Safety Related Equipment; Revision 8

NP 8.4.10; Exclusion of Foreign Material from Plant Components and Systems; Revision 17

OI 62A; Motor-Driven Auxiliary Feedwater System (P-38A and P-38B); Revision 28

OI 62B; Turbine-Driven Auxiliary Feedwater System (P-29); Revision 12

OI 70; Service Water Operation; Revision 53

OI 100; Adjusting SI Accumulator Level and Pressure; Revision 27

OI 128; SI System Fill and Vent Unit 1; Revision 12

OI 135; Fill and Vent the RHR System Unit 1; Revision 4

OI 135A; Fill and Vent Train A RHR System Unit 1; Revision 9

OI 135B; Fill and Vent Train B RHR System Unit 1; Revision 11

OI 168; Emergency Diesel Generator Operability; Revision 2

OM 4.3.2; EOP/AOP Verification/Validation Process; Revision 12

PBF 2139; SI Accumulator Leak Rate Worksheet; Revision 1

RMP 9043-31 Emergency Diesel Generator G-03 2 Year Electrical Inspecion; Revision 7

RMP-9326; General Maintenance Inspection of Check Valves; Revision 2

SEP-2.1 Unit 1; Shutdown LOCA with RHR Aligned for Low Head Injection; Revision 13

SEP-2.2 Unit 1; Shutdown LOCA with RHR Aligned for Decay Heat Removal; Revision 13

SEP-2.3 Unit 1; Cold Shutdown LOCA; Revision 12

1TS-ECCS-002; Safeguards System Venting (Monthly) Unit 1; Revision 6

TS 81; Emergency Diesel Generator G-01 Monthly; Revision 72

TS-81; Emergency Diesel Generator, G-01; Revision 72, Test date August 6, 2006

Surveillances (completed)

0-PT-AF-1; P-38A Motor-Driven AFW Pump Backup Air System Pressure Decay Test; dated June 24, 2006

0-PT-AF-2; P-38B Motor-Driven AFW Pump Backup Air System Pressure Decay Test; dated June 23, 2006

1-PT-AF-3; 1P-29 Turbine Driven AFW Pump Backup Air System Pressure Decay Test (Refueling) Unit 1;dated October 8, 2005,

2-PT-AF-3; 2P-29 Turbine Driven AFW Pump Backup Air System Pressure Decay Test (Refueling) Unit 2; dated April 15, 2005, June 22, 2006

IT 01; High Head Safety Injection Pumps and Valves (Quarterly) Unit 1; dated July18,2006

IT 01A; High Head Safety Injection Pumps and Valves (Cold Shutdown) Unit 1; dated November 2, 2005

IT-8A; Cold Start of Turbine Driven Auxiliary Feedwater Pump and Valve Test (Quarterly) Unit 1; dated June 22, 2006

IT-8B; TDAFWP Suction from SW MOV Exercise Test Unit 1;dated November 1, 2005 and June 23, 2006

IT-9A; Cold Start of Turbine Driven Auxiliary Feedwater Pump and Valve Test (Quarterly) Unit 2; dated June 30, 2006

IT-9B; TDAFWP Suction from SW MOV Exercise Test Unit 2;dated October 23, 2005, dated June 30, 2006

IT-10; Test of Electrically-Driven Auxiliary Feedwater Pumps and Valves; dated June 22, 2006

PC-73; Auxiliary Feedwater Pump Bearing Cooling Flush (Annual); dated January 28, 2006

Work Orders

WO0201903; Inspect North Zurn Strainer;dated March 7, 2005

WO0209043; Inspect South Zurn Strainer; dated June 9, 2006

WO0216243; Open and Inspect Check Valve FP-00296A; dated October 10, 2003

WO0219247; Open and Inspect Check Valve FP-00304A; dated July 20, 2004

WO0412087; Open and Inspect Check Valve SW-0035A; dated December 12, 2005

WO9819093; SW-2816 Preventive Maintenance; dated June 7, 2002

WO9934090; 1AF-4001 Preventive Maintenance; dated May 2, 2001

LIST OF ACRONYMS USED

AAC	Alternate AC
AC	Alternating Current
ADAMS	Agencywide Documents Access and Management System
AFP	Auxiliary Feedwater Pump
AFW	Auxiliary Feedwater
ASME	American Society of Mechanical Engineers
CDBI	Component Design Bases Inspection
CCW	Component Cooling Water
CFC	Containment Fan Cooler
CFR	Code of Federal Regulations
CST	Condensate Storage Tank
DC	Direct Current
DRS	Division of Reactor Safety
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
ESF	Engineered Safety Feature
FSAR	Final Safety Analysis Report
GL	Generic Letter
HZ	Hertz
IEEE	Institute of Electrical and Electronics Engineers
IPEEE	Individual Plant External Event Evaluation
IMC	Inspection Manual Chapter
IST	Inservice Testing
LCO	Limiting Condition for Operability
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
MCC	Motor Control Center
MDAFP	Motor Driven Auxiliary Feedwater Pump
MNGP	Monticello Nuclear Generating Plant
MOV	Motor-Operated Valve
NCV	Non-Cited Violation
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NUMARC	Nuclear Management and Resources Council
OA	Other Activities
PARS	Publicly Available Records
PRA	Probabilistic Risk Assessment
RHR	Residual Heat Removal
RIS	Regulatory Information Summary
RWST	Refueling Water Storage Tank
SBO	Station Blackout
SDP	Significance Determination Process
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SPAR	Standardized Plant Analysis Risk
SQUG	Seismic Qualification User Guidelines
SW	Service Water

TDAFP	Turbine Driven Auxiliary Feedwater Pump
TS	Technical Specifications
URI	Unresolved Item
V	Volt
VDC	Volt Direct Current